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JRBuchanan

August 2, 1976

Docket No.: 50-271

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. 25 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS). The amendment consists of changes to the Technical Specifications in response to your application dated April 23, 1976, as supplemented May 25, 1976, and staff discussions.

This amendment modifies the Technical Specifications relating to the replacement of 136 of 368 fuel assemblies in the reactor core of VYNPS constituting refueling of the core for cycle 4 operation.

In order to facilitate future reviews in which Lead Test Assemblies are used in core reloads, please submit the results of your findings concerning the use of LTA's.

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

Sincerely,

Original Signed by

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

Enclosures:
1. Amendment No. 25
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

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

August 2, 1976

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cc w/enclosures: See next page

Yankee Atomic Electric Company

cc: w/enclosure

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cc w/enclosures and copy of
VY's filings dtd 4/23/76 &
5/25/76

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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. 25
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated April 23, 1976, as supplemented May 25, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: August 2, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
12-b & 12-c	12-b & 12-c
13	13
14-a & 14-b	14-a & 14-b
15	15
47	47
64	64 & 64a
65	65
71	71
77 & 78	77 & 78
180-a	180-a
180-c thru 180-f	180-c thru 180-f
180-h thru 180-k	180-h thru 180-k
180-n	180-n1 thru 180-n3

The changed areas on the revised pages are shown by marginal lines.

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
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
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	3293 MWt
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
R-Factor	(7 x 7) 1.098 (8 x 8) 1.10
Core Bypass Flow Rate	10.25 Mlb/hr
Core Inlet Temperature	527.68°F

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 conditions 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.

The transient results provide the maximum reduction in Critical Power Ratio (ΔCPR) which is then added to the fuel cladding integrity safety limit MCPR to provide a conservative operating MCPR limit (Specification 3.11C).

The 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.

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.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR (Specification 3.11C).

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

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 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 station 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 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 reduced APRM 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 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 the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

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 the fuel cladding integrity safety limit. 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 the specified values, thus preserving the APRM rod block safety margin.

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.

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 $\leq$10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit 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.

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 therfore adequate margin exists.

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

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		≤5 x 10^5 cps (Note 3)
2	b. Detector not Fully Inserted	X	X		
	Intermediate Range Monitor				
2	a. Upscale	X	X		≤108/125 full scale
2	b. Downscale (Note 4)	X	X		≥5/125 full scale
2	c. Detector not Fully Inserted	X	X		
	Average Power Range Monitor				
2	a. Upscale (Flow Bias)			X	≤0.66W + 42% (Note 5)
2	b. Downscale			X	≥2/125 full scale
	Rod Block Monitor (Note 6)				
1	a. Upscale (Flow Bias) (Note 7)			X	≤0.66W + 41% (Note 5)
1	b. Downscale (Note 7)			X	≥2/125 full scale
1	Trip System Logic	X	X	X	
1	Scram Discharge Volume	X	X	X	≤12 gallons

3.2 (Continued)

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 1295oF; 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.

3.2 (Continued)

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease below the fuel cladding integrity safety limit. 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.

3.2 (Continued)

The APRM rod block trip is flow referenced and prevents a singificant 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 the fuel cladding integrity safety limit.

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 the fuel cladding integrity safety limit.

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 capability 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).

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

   (c) The operating power level shall be limited so that the MCPR will remain above the fuel cladding integrity safety limit 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.

3.3 (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 functions. 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 the fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RMB 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.

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 greater than the fuel cladding integrity safety limit.

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.

LIMITING CONDITIONS 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 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 values shown in Figures 3.11-1A through E. 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 conditions 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.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied 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.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, the Operating MCPR Limit shall be $\geq$1.20 for 7 x 7 fuel and $\geq$1.21 for 8 x 8 fuel 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 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)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, 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 10CFR50, Appendix K limit. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10CFR50. A complete discussion of each code employed in the analysis is presented in Reference 1.

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 10CFR50 Appendix K, NEDO-20566, January 1976.

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

Fuel Parameters:

Fuel Type	Fuel Bundle Geometry	Peak Linear Heat Generation Rate (Kw/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
Generic B (Reload 1)	7 x 7	NA*	1.5	1.18
8D219 (Reload 2)	8 x 8	NA	1.5	1.18
8D274 (Reload 3)	8 x 8	NA	1.5	1.18
LTA (Reload 3)	8 x 8	NA	1.5	1.18
High Gd_2O_3 (Reload 3)	8 x 8	NA	1.5	1.18

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

*Fuel is peak cladding temperature limited, therefore, fuel cannot operate at peak LHGR due to MAPLHGR restrictions.

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, 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.

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.

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

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].

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 Coolant 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 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 for the same postulated transient event.

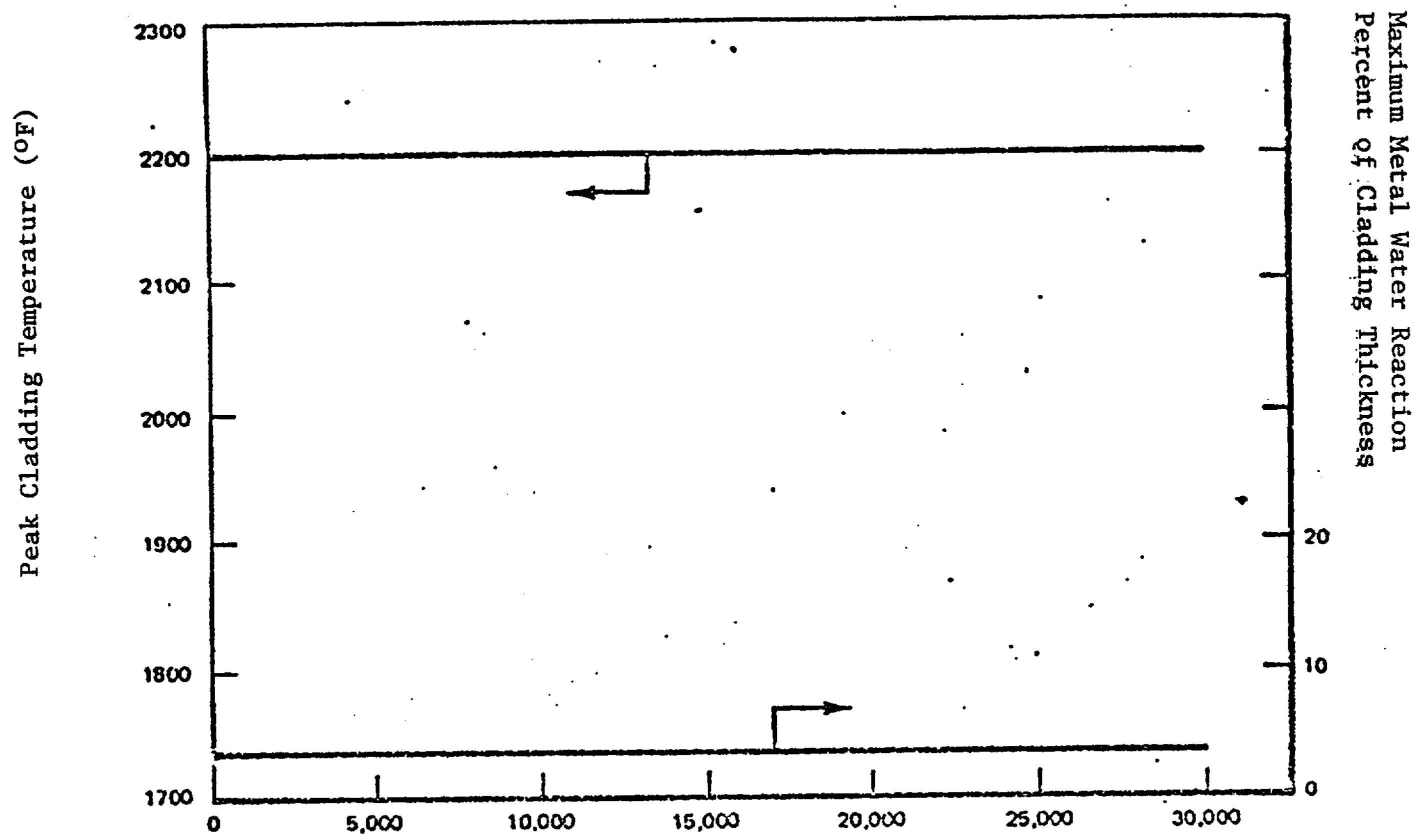
The K_f factor curves shown in Figure 3.11.2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

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.

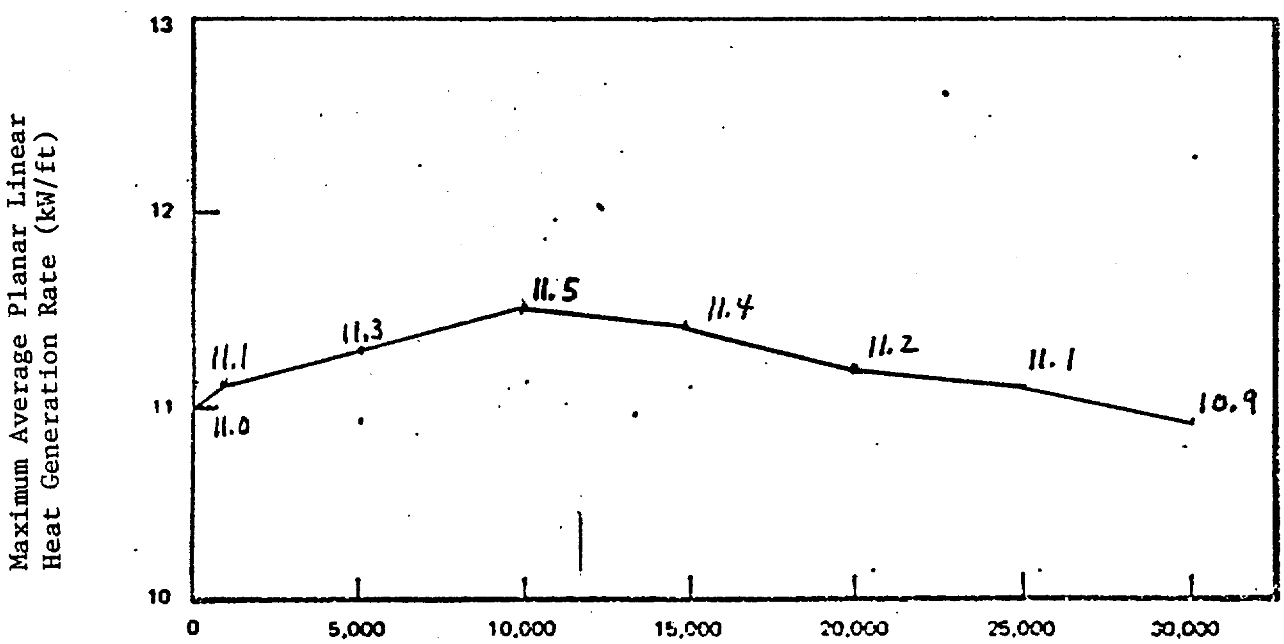
3.11 (Continued)

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.

Figure 3.11-1C
Vermont Yankee Bypass Flow Holes Plugged, 8 x 8, 8D274 Fuel

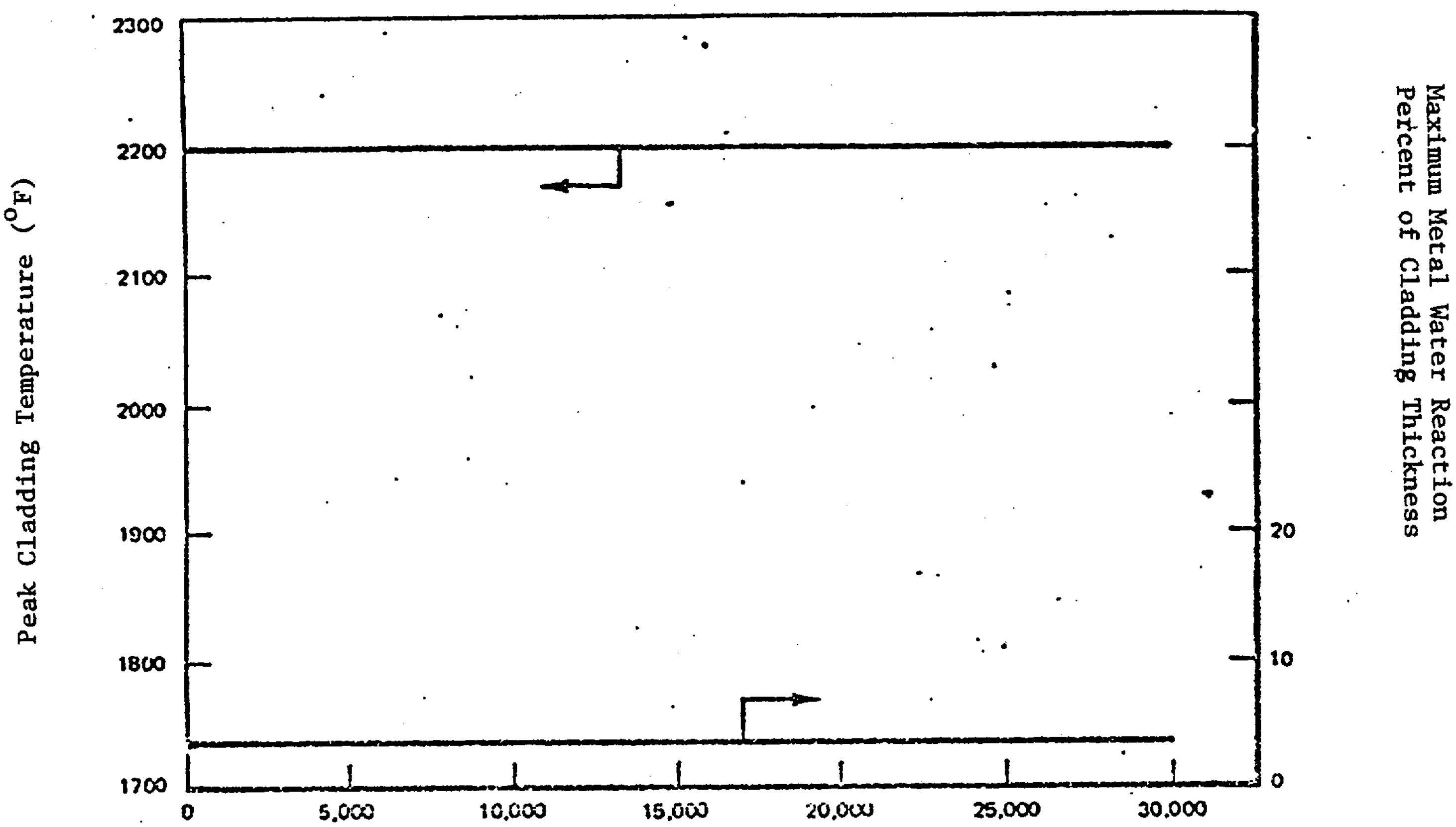


Peak Cladding Temperature versus Planar Average Exposure

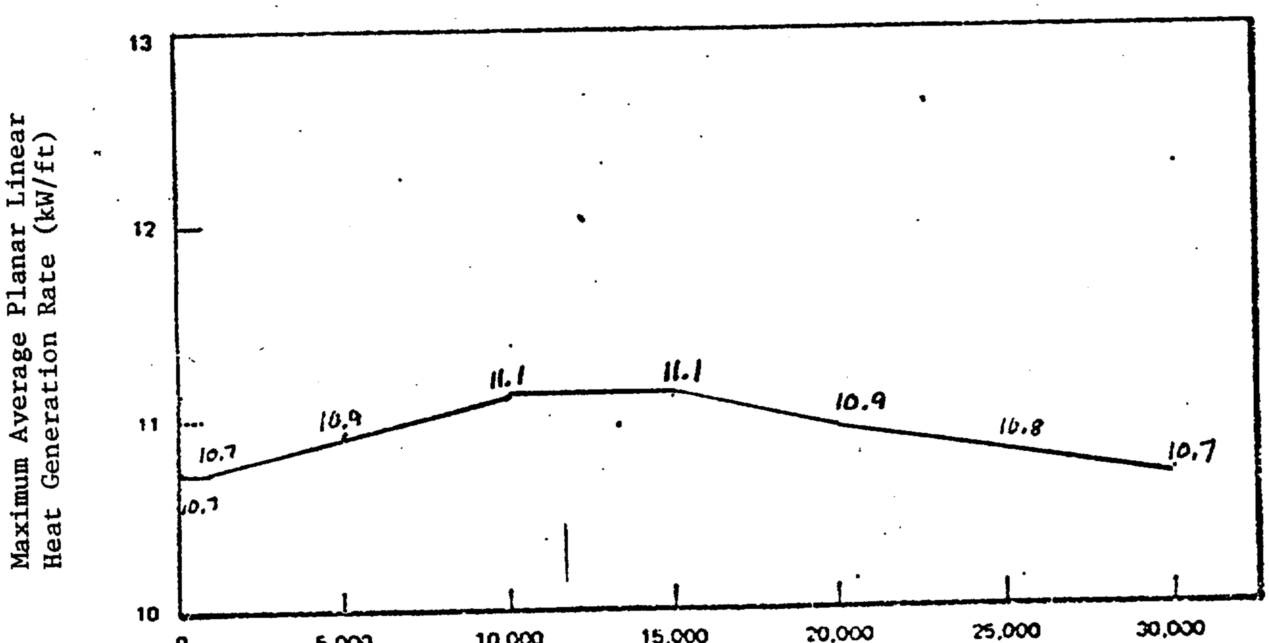


Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Planar Average Exposure

Figure 3.11-1D
Vermont Yankee Bypass Flow Holes Plugged, 8 x 8, LTA

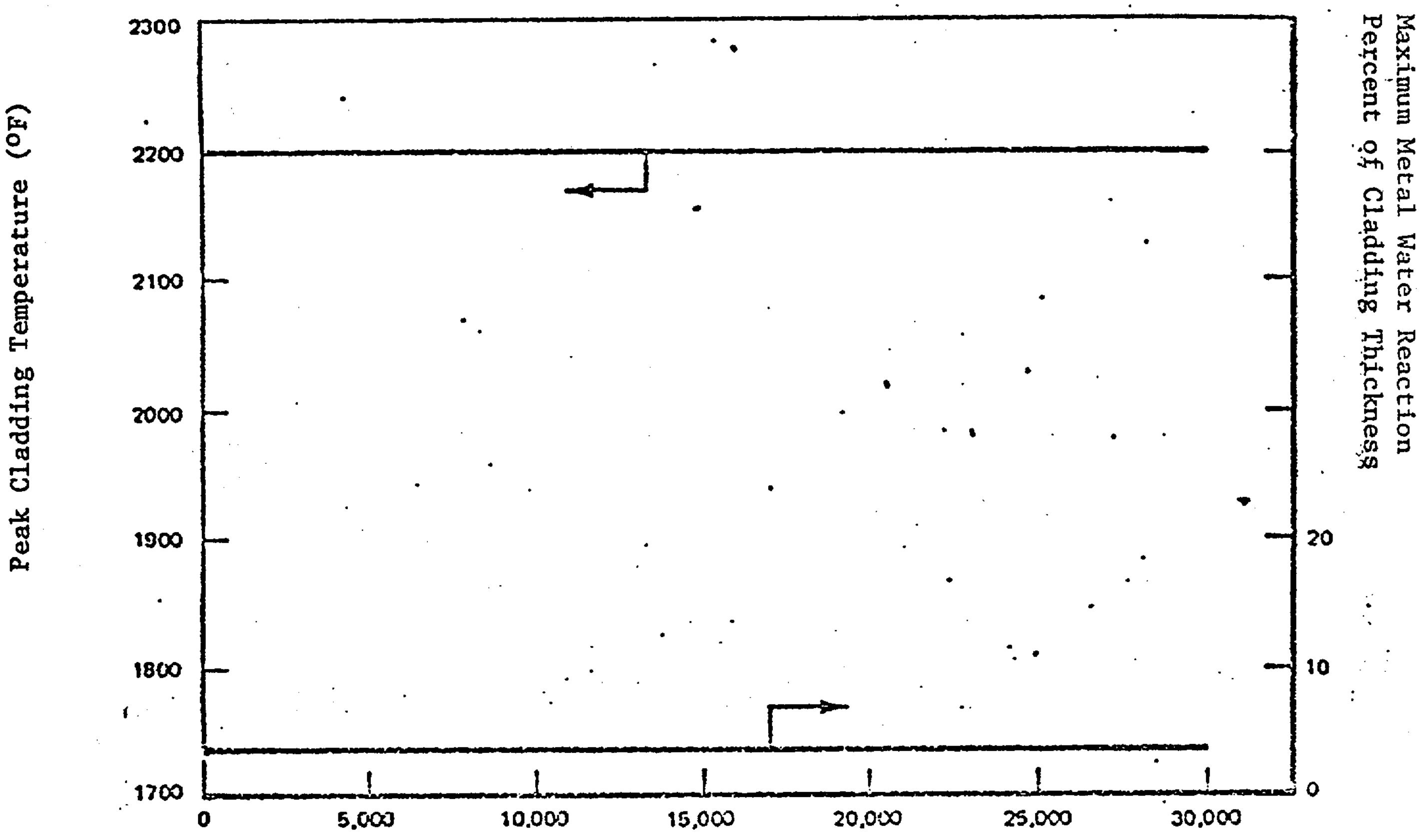


Planar Average Exposure (MWd/t)
PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE

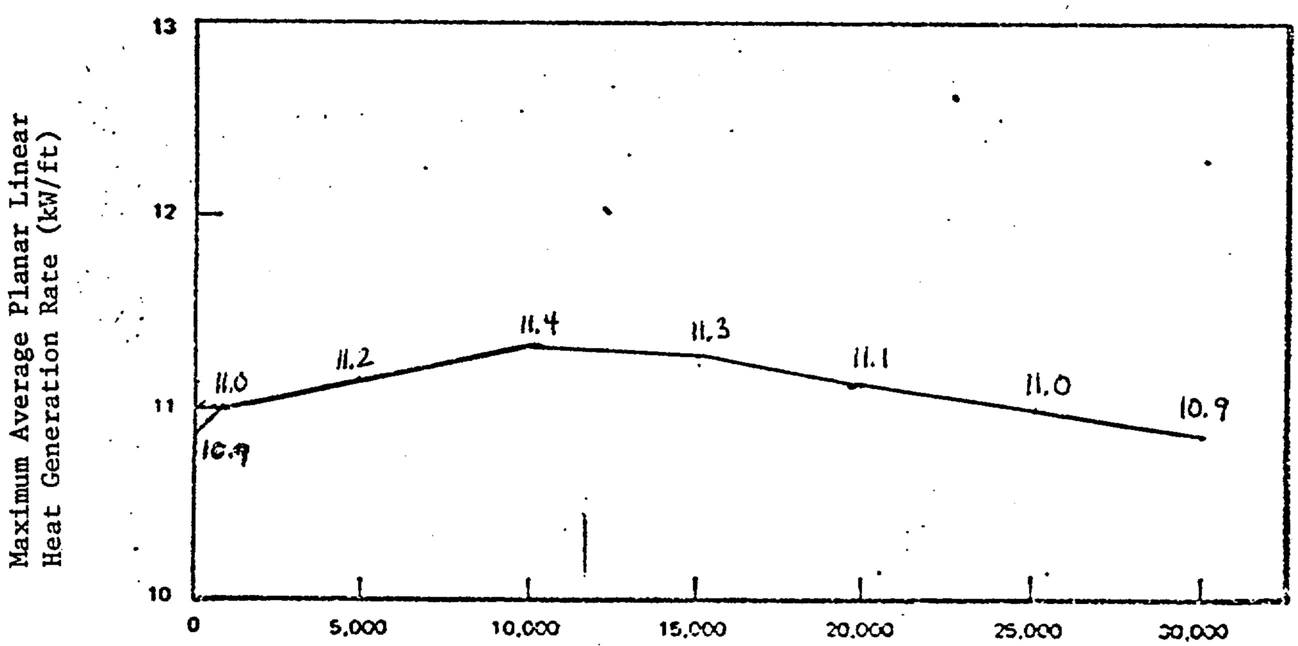


Planar Average Exposure (MWd/t)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

Figure 3.11-1E
Vermont Yankee Bypass Flow Holes Plugged, 8 x 8, High Gd_2O_3 Assembly



Planar Average Exposure (MWd/t)
Peak Cladding Temperature versus Planar Average Exposure



Planar Average Exposure (MWd/t)
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Planar Average Exposure

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO FACILITY LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated April 23, 1976, and supplemented by letter dated May 25, 1976, Vermont Yankee Nuclear Power Corporation (VYNPC) requested an amendment to Facility Operating License No. DPR-28. The amendment would modify the Technical Specifications for the Vermont Yankee Nuclear Power Station (VYNPS) to permit operation of the facility with:

1. Up to 130 General Electric (GE) 8 x 8 reload assemblies
2. Two Lead Test Assemblies (LTA's)
3. Four assemblies with higher gadolinia content than the standard reload assemblies
4. All 136 reload assemblies with two 9/32 inch holes drilled in the lower tie plates
5. A modification to the Rod Block Monitor setpoint

2.0 BACKGROUND

VYNPC has proposed to reload the VYNPS reactor with up to 134 GE 8 x 8 assemblies with an average enrichment of 2.74 weight percent (wt %) uranium-235 with varying gadolinia content. Two lead test assemblies with 8 x 8 fuel rod configuration but different rod design will also be loaded for cycle 4 (Reload 3) operation.

The 136 new fuel assemblies will replace 136 8 x 8 fuel assemblies having an average enrichment of 2.19 wt % U-235. Cycle 4 operation will be with a core mixture of 40 7 x 7 with 2.3 wt % U-235, 192 8 x 8 with 2.19 wt % U-235, 134 8 x 8 with 2.74 wt % U-235, and 2 8 x 8 Lead Test Assemblies with 2.60 wt % U-235. Table 1 lists the fuel type and the number of assemblies in both cycle 3 and cycle 4.

The documentation submitted in support of the proposed reload relates to the GE Boiling Water Reactor reload licensing application for 8 x 8 fuel (references 2 and 3), the Lead Test Assemblies (references 2 and 4), high gadolinia content assemblies (references 1, 2, and 3), drilled reload assemblies (references 1 and 10). In addition to the referenced material VYNPC, in response to our request, provided additional information by letters dated June 23, July 6, and July 19, 1976.

The proposed changes to the VYNPS Technical Specifications and bases include:

1. A change in the bypass void effect on TIP (traversing incore probes) from 3.53 to 3.97% (section 3.3).

2. A change in the operating limit minimum critical power ratio (MCPR) to 1.21 for 8 x 8 fuel and 1.20 for 7 x 7 fuel. The operating limit MCPR for the previous cycle was 1.28 for both 8 x 8 and 7 x 7 fuel (section 3.4.1).

3. A change in the Rod Block monitor trip setting from $\leq$ 0.66W + 40% to $\leq$ 0.66W + 41% (where W is the fraction of full flow) (section 3.4.3).

4. A change to eliminate the subtraction of the fuel densification power spike penalty when setting the maximum operating linear heat generation rate (section 6.0).

Each of these proposed Technical Specification changes is discussed in the section of this report indicated in parentheses above.

TABLE 1

Fuel Type and Number

Fuel Type	Cycle 3 Number	Cycle 4 Number
Reload 1 7D230	40	40
Reload 2 8D219	328	192
Reload 3		
8D274H	0	44
8D274L	0	86
LTA	0	2
High Gd_2O_3	0	4
Total	368	368

3.0 EVALUATION

3.1 Nuclear Characteristics

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (reference 2). Although later supplements to this report are undergoing review by the staff, this topical report is applicable for use for reactors containing 8 x 8 reload fuel. Up to 134 GE 8 x 8 reload fuel bundles with an average enrichment of 2.74 wt % U-235 will be loaded throughout the core. Forty-four of the reload fuel assemblies have high gadolinia content (8D274H) and eighty-six have a low gadolinia content (8D274L). Four assemblies, as shown in Table 1 will contain rods of higher gadolinia content than the 8D274H rods (high Gd_2O_3). The assembly design is identical to that of the other reload assemblies. The four high gadolinia content assemblies will be placed in the core for cycle 4 rather than a lesser number to assure symmetry. In addition, two Lead Test Assemblies (LTA's) will be loaded and are expected to be operated for four full reactor cycles.

The LTA's have a total fueled length of 150 inches compared to 144 inches for a conventional 8 x 8 assembly. The top six inches and the bottom four inches in each LTA contain natural uranium pellets. The remaining 140 inches contain enriched uranium. The average enrichment of each LTA including the natural uranium (10.0 inches per rod) is 2.60 percent. The core contains a total of 368 fuel assemblies. Thus, about 37% of the fuel assemblies are being replaced for the reload.

The loading pattern consists of 8 x 8 reload assemblies in a symmetrical pattern throughout the core. The two LTA's are located in a symmetrical array towards the periphery of the core. The high gadolinia reload assemblies are loaded in the interior of the core while the low gadolinia reload assemblies are loaded at the outer portions of the core. The data in reference 1 indicate that the nuclear characteristics of the reload 3 core (including the two LTA's) are similar to the previous core. Thus, the total control system worth, and the temperature behavior of the reconstituted core will not differ significantly from those values previously reported for VYNPS.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25% ΔK subcritical in the most reactive operating state with the most reactive rod fully withdrawn and with all the others fully inserted. For cycle 4 a minimum shutdown margin of 0.0104 ΔK was calculated.

The 136 fuel assemblies loaded for cycle 4 will have drilled lower tie plates which will increase core bypass flow. As a result of the new flow configuration caused by the partial drilling, the void coefficient of reactivity has been increased from -11.6×10^{-4} to -10.33×10^{-4} at a 39.7% core average void fraction. The significance of this change on the core transient analysis is discussed in section 3.3. For conservatism the void coefficient contains a multiplier of 1.25

The upper limit on Doppler coefficient changes from a value of -1.07×10^{-5} to a value of -1.226×10^{-5} for cycle 4 and conservatively contains a .9 multiplier at the most reactive condition.

Information presented in reference 1 indicates that a boron concentration of 800 PPM in the moderator will bring the reactor subcritical by 0.05 Δk at 20°C, Xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The full power scram reactivity curve used for reload 3 is shown in Figure 7-6 of reference 1. The scram curve includes a 0.8 multiplier on the reactivity for conservatism.

The use of natural uranium in the extremities of the LTA is intended to provide a more efficient use of the total amount of uranium in the core. The nuclear characteristics of the LTA's are not significantly different from the standard 8 x 8 8D274L fuel assemblies.

Thus, based on the information presented in VYNPC application (reference 1) and supplemented by the generic 8 x 8 reload report (reference 2) which is still undergoing review, the nuclear characteristics and performance of the reconstituted core for cycle 4 are acceptable.

3.2 Mechanical Design

The two types of reload 3 fuel have the same mechanical configuration and fuel assembly enrichments as the 8D274L and 8D274H fuel assemblies described in the 8 x 8 generic reload report, except that holes are drilled in the lower tie plate. Two 9/32 inch holes are drilled in the lower tie plate of the reload assemblies to provide bypass flow. The channel wall thickness for the reload assemblies is the same as the standard product line fuel channels (nominal 0.080 inch wall thickness).

The two lead test assemblies are similar in design to those loaded earlier into the Peach Bottom Unit No. 2 reactor and are similar in outward design to the reload assemblies. Table 2 gives a comparison (repeated from reference 1) of the lead test assembly with the 8D274L design.

The lead test assembly consists of 62 fuel rods and two water rods, one of which is also a spacer positioning rod. The fuel rods are composed of 95% theoretical density UO_2 pellets stacked in a Zircaloy cladding tube which is evacuated and backfilled with helium, as are the standard fuel rods.

The standard fuel rods contain a column of enriched UO_2 144 inches in length with a plenum length of 10 inches. The lead test assembly fuel rods contain a column of UO_2 150 inches long. Of this length 140 inches consist of enriched UO_2, 10 inches (6 on top, 4 on bottom) consist of natural UO_2. Except for the fact that the power generation will be less in the natural uranium portions of the fuel rod, the behavior of the naturally enriched UO_2 would not be expected to be different from that of the more highly enriched UO_2.

On the basis of our review of the generic 8 x 8 reload report (2) and the reload submittal (1) we conclude that the design of the fuel proposed for cycle 4 operation at Vermont Yankee is acceptable.

TABLE 2

8D274L Reload Fuel Assembly and LTA Design Specifications

	8D274L	LTA
FUEL ASSEMBLY		
Geometry	8x8	8x8
Rod Pitch (in.)	0.640	0.640
Water to Fuel Volume Ratio	2.60	2.75
Heat Transfer Area (ft^2)	97.6	98.0
Weight of UO_2 (kg)	208.0	207.1
Weight of U(kg)	183.4	182.6
Average Enrichment (w/o U-235)	2.74	2.60*
Finger Springs	Yes	Yes
FUEL RODS		
Active Fuel Length (in.)	144.0	150.0*
Gas Plenum Length (in.)	16	9.5
Fill Gas	Helium	Helium
Getter	Yes	Yes
FUEL		
Material	Sintered UO_2	Sintered UO_2
Pellet Diameter (in.)	0.416	0.410
Pellet Length (in.)	0.420	0.410
Pellet Immersion Density (% TD)	95.0	95.0
CLADDING		
Material	Zr-2	Zr-2
Outside Diameter (in.)	0.493	0.483
Thickness	0.034	0.032
WATER ROD		
Material	Zr-2	Zr-2
Outside Diameter (in.)	0.493	0.591
Thickness	0.034	0.025
SPACERS		
Material	Zr-4 with Inconel X-750 Springs	Zr-4 with Inconel X-750 Springs
Number per bundle	7	7
FUEL CHANNEL		
Material	Zr-4	Zr-4
Outside Dimension (in.)	5.278	5.278
Wall Thickness (in.)	0.080	0.080

*Includes natural Uranium (10.0 inches per rod).

On the basis of similarity to the generic design, the limited number of assemblies in the core (two) and the information presented in the reload submittal (reference 1), we conclude that it is acceptable to operate Vermont Yankee for cycle 4 with the two lead test assemblies described in references 2 and 4.

As stated earlier and as shown in Table 1, four fuel assemblies will contain rods of higher gadolinia content than the 8D274H rods. The design is identical to that of the other reload assemblies. Although, VYNPS has a linear heat generation rate (LHGR) limit of 13.4 kw/ft, the mechanical design of the fuel rods was done assuming a LHGR of 14.4 kw/ft for conservatism.

We have reviewed the design of these high gadolinia assemblies and find that their use for cycle 4 of Vermont Yankee is acceptable.

3.3 Thermal-Hydraulics

The GE 8 x 8 fuel reload topical (2) and the GE/BWR Thermal Analysis Basis (GETAB) (7) are referenced to provide a basis for the core thermal hydraulics analysis for cycle 4.

A significant part of the review of the proposed cycle 4 operation dealt with the question of the acceptability of the flow distribution and consequent thermal margin resulting from the loading of reload fuel with drilled lower tie plates into the plugged core. The concern was that the inlet bypass flow might be nonuniform and that this, combined with nonuniform heat generation in the fuel bundles might cause more voiding in the hotter bypass regions than had been accounted for. The bypass voiding is accounted for as an uncertainty in the determination of the Safety Limit MCPR.

In the staff Safety Evaluation Report on Modification to Eliminate Significant In-Core Vibration (in BWR's) the staff reviewed and found acceptable operation with both the completely drilled and completely plugged cores. Regarding partially drilled cores the staff's position was:

"For those reactors in which the 1-inch bypass flow holes are plugged but not all fuel bundles are drilled we conclude that the outreactor flow test sufficiently demonstrated that the modification will reduce significantly in-core tube vibration and hence channel box damage. However, the allowable power level after such modifications must be reviewed individually for each reactor considering normal operation, anticipated transients and accidents (NEDE 21156)."

Vermont Yankee is the first BWR to change the technical specifications to an MCPR lower than the fully plugged value based on the benefit of partial drilling.

In order to demonstrate that the inlet bypass flow was uniform, the licensee referenced three experiments done by General Electric at their 32-bundle test facility. In one of these tests (1603) all the fuel bundles had drilled lower tie plates. A pitot tube was used to measure the velocity profile. With all the bundles drilled the flow distribution was relatively flat at both low and high axial elevations along the fuel bundle. In tests 1913 and 1914 in the 32-bundle test assembly, some bundles were drilled and the rest of the test assembly was plugged. Accelerometer readings demonstrated that significant vibration of the instrument tubes was eliminated. No video information or pitot tube readings were recorded.

The amount of bypass voiding is calculated using a core hydraulics model which calculates the flow distribution in both the bundles and the bypass region assuming a given core pressure drop and total flow rate. The bypass voiding is conservatively calculated by assuming no mixing with bypass flow from cooler regions. In the transient analyses the bypass flow is assumed to be at the inlet enthalpy until the bypass flow is mixed in the exit plenum. While this assumption is not in the conservative direction for the case of bypass voiding, the effect on the transient analysis would not be significant since the bypass flow is a small fraction of the total core flow and the void fraction used in the transient analysis is a core average value.

The bypass voiding will affect the power distribution in the bundle and this will have an effect on the R factor used in the GEXL correlation as well as the uncertainty in the R factor, which is used in calculating the Safety Limit MCPR.

In generic discussions with General Electric, General Electric has stated that these effects on the R factor are considered and are within the conservatism used in obtaining the uncertainty in the R factor. We conclude this to be acceptable.

We agree that the thermal hydraulics performance with bypass voiding as a result of partial loading with drilled fuel has been conservatively considered for cycle 4 operation for VYNPS.

3.4 Accident and Transient Analysis

3.4.1 Anticipated Transients

The licensee, in reference (1) has stated that "all transients which are the basis of the existing license were reviewed, and those transients which have been limiting in the past with respect to safety margins and are significantly sensitive to the core transient parameter deviations were reanalyzed." The methods used for these analyses are described in references (5) and (6). The transients reanalyzed for cycle 4 were the Turbine Trip Without Bypass and the Loss of Feedwater Heater. The results of these reanalyses are given in Table 7.3 of reference (1). The highest ΔMCPR occurs for the 8 x 8 fuel during the Tubine Trip Without Bypass Transient. The value obtained was 0.15 at the end of cycle 4. This ΔMCPR is less than that obtained for cycle 3 of .22. Thus, because the safety limit MCPR was the same for both analyses (MCPR=1.06) Vermont Yankee requsted that the operating limit MCPR be reduced from 1.28 to 1.21 in the Vermont Yankee technical specifications.

The MCPR serves to protect the fuel against excessive clad temperatures in the event of an abnormal operating transient. The proposed reduction in ΔMCPR is attributed to the improved transient performance resulting from the elimination of voiding in the bypass region by use of fuel with drilled bypass holes in the lower tie plate assembly. Based on our review of the referenced material and because the new ΔMCPR prevents reaching the safety limit in the event of all transients (as did the old ΔMCPR), we conclude that the change is acceptable.

3.4.2 Overpressure Analysis

In reference 1 VYNPC presented the results of an overpressure analysis 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 with high neutron flux scram. The analysis was performed for 104.5% power with the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for the relief function of the safety/relief valves, with all safety valves operative. The results of this analysis indicate that the peak pressure at the vessel bottom would be 1279 psig. Furthermore, generic analysis applied to VYNPS showed that for the aforementioned overpressure event, the failure of one safety valve would cause the maximum vessel pressure to increase by 20 psig. Hence, the maximum peak pressure at the vessel bottom for MSIV closure with flux scram, no relief function of the safety relief valves and one failed safety valve is calculated to be 1299 psig; this results in about a 76 psi margin below the code allowable, which is acceptable to us.

3.4.3 Rod Withdrawal Error

The licensee has analyzed the Rod Withdrawal Error according to the assumptions given in reference (1). The results show that a rod block monitor (RBM) set point of 107% will stop rod withdrawal when the critical power ratio is 1.10 which is greater than the 1.06 MCPR safety limit. Based on this analysis, the proposed Rod Block Monitor set point of $\leq$ 0.66W + 41% (where W is the fraction of full coolant flow) is acceptable.

3.4.4 Loss-of-Coolant Accident

The licensee has used the cycle 3 analysis of the Loss-of-Coolant Accident for cycle 4. This analysis was done using the assumption that the core plate is completely plugged. The analysis is presented in reference (8) and was done according to Appendix K to 10 CFR Part 50. Our evaluation of this analysis is given in reference (9). We find this to be a conservative approach since the reload fuel will provide additional bypass flow which will reduce the core reflood times in the LOCA event, and is therefore acceptable.

The response of the lead test assemblies and the high gadolinia assemblies to a Loss-of-Coolant accident was analyzed by the licensee using the same assumptions as for the 8D274 reload fuel. The MAPLHGR curve is given in Table A6.1 of reference (1). Since the LTA's are of similar nuclear design to the standard assemblies, we conclude that the previous LOCA analysis is applicable and is therefore acceptable.

3.4.5 Main Steam Line Break, Refueling Accident, Control Rod Drop Accident

The analyses of the following accidents were listed by the licensee as being covered by the generic analyses give in reference (2).

- Main Steam Line Break Accident
- Refueling Accident
- Control Rod Drop Accident

Based on our previous review of the referenced material for VYNPS (9), we conclude that the results provided by the generic analyses are acceptable.

3.4.6 Loading Error Accident

The following assumptions are made for this accident:

A. A reload bundle is rotated 180 degrees in a location near the center of the core or a bundle is inserted in an improper location; and

B. The error is not discovered in the subsequent core verification and the reactor is operated.

For Vermont Yankee, the case of the fuel bundle inserted in an improper location gave the worst results. For this case the peak linear heat generation rate is 18.1 kw/ft and the minimum critical power ratio is 0.91 in the misplaced fuel bundle (adjacent fuel bundles are not affected).

Since this accident results in a CPR of less than the safety limit, it is expected that some of the fuel rods in the bundle will experience boiling transition and must therefore be presumed to fail.

Detection of any abnormal fuel degradation is accomplished in the Vermont Yankee facility by measurement of reactor coolant radioactivity levels, measurement of off-gas radioactivity levels at the air ejector and measurement of radiation levels in the main steam tunnel at the main steam line isolation valves.

Sampling of the coolant for radio-iodine is required by Technical Specifications if a change in the off-gas activity of 25% or 5000μCi/sec (whichever is greater) is detected.

The allowable limit for iodine in the reactor coolant, 1.1μCi/gm dose equivalent I-131, approximately corresponds to the levels expected immediately after gross failure of two fuel pins. If the failure of a large number of fuel pins (in the order of 80) causes the off-gas activity to increase above 0.3 Ci/sec (30-minute decay value) for more than 15 minutes, the air ejector would be automatically tripped, resulting in shutdown of the reactor. Similarly, if the off-gas activity level increases above 1.5 Ci/sec (30-minute decay value) for more than 1 minute, the air ejector would be automatically tripped. This level would be exceeded under post-startup conditions if a few gross failures of fuel pins occurred sequentially and may be exceeded for a gross failure of a single pin in some cases. The third indication, alarm or closure of the main steam line isolation valves of the radiation monitors, would occur at 1.5 and 3 times the background radiation levels (caused mainly by short-lived N-16). These set points correspond to the levels that would result from failure of several fuel pins.

Fresh fuel would have a smaller radioactive inventory and would be less likely to exceed the limits discussed above during the first few weeks of operation. The potential offsite radiological consequences would be less for this case, however.

We conclude that the existing technical specifications for the Vermont Yankee reactor provide assurance that significant abnormal fuel degradation, including that which might result from an undetected fuel loading error, would be detected and reported to the NRC and that reactor shutdown would automatically result in the event that large numbers of fuel pins experienced gross failure.

Any radioactivity which passed the main steam line isolation valves and air ejectors prior to their closures would be retained on the charcoal beds of the off-gas treatment system where it would decay to levels at which significant offsite exposures would not result. Even in the unlikely event that the activity collected on the charcoal beds were released by some unrelated independent event, the resultant offsite exposures would be well within the guidelines of 10 CFR Part 100.

In addition to the detection capabilities and Technical Specification requirements, VYNPC augmented their normal Quailty Assurance procedures for verifying fuel position and independently and separately verified that each fuel assembly was loaded into the correct position in its proper geometry. Inspectors from the NRC Office of Inspection and Enforcement visited the reactor site and independent of VYNPC inspected the core and verified that the fuel was correctly placed in the core. The inspection and verification was completed July 23, 1976; no assemblies were found to be mislocated or misoriented.

Thus, based on the fact that the failure of the fuel can be detected and the augmented surveillance by both VYNPC and the NRC Office of Inspection and Enforcement, we consider the consequences of a calculated CPR of 0.91 for a Fuel Loading Error Accident at Vermont Yankee to be acceptable.

4.0 PHYSICS STARTUP TESTING

As part of our review of Reload 3, VYNPC was requested to provide a description of the Physics Startup testing program. In response to our request, this program was described in VYNPC letter dated June 23, 1976. The results of this test will be reported in the next VYNPS operating report. We find the Startup Physics testing program and reporting schedule acceptable.

5.0 SURVEILLANCE OF CHANNEL BOX WEAR

Because the partially drilled core configuration is new, we asked Vermont Yankee to institute a surveillance program to verify that that the new flow configuration would not produce vibrations of the instrument tubes which could lead to damage of the channel boxes. Vermont Yankee has proposed the following surveillance program.

A full set of Traversing Incore Probe (TIP) traces will be obtained on a normal schedule of every two weeks following plant startup.

Accelerometers have been placed on the following four Local Power Range Monitor (LPRM) tubes:

LPRM Locations	# of Drilled Fuel Assemblies Surrounding the LPRM String
24-25	0
32-25	1
16-33	2
24-41	3

Figure 3-2 of reference (1) gives the orientation of drilled fuel assemblies with respect to LPRM locations. The accelerometers will be monitored monthly in conjunction with the TIP traces. VYNPC has stated that every effort will be made to have the system operational at the time of plant startup following the refuel outage; however, due to prior scheduling and manpower commitments it may not be possible. In any event it is presently anticipated that the monitoring system will be operational no later than 30 days following plant startup.

The data will be collected until the next refueling outage. At that time, the data will be correlated with visual observations of the fuel channels surrounding the monitored LPRM tubes. After such a correlation has been made, a decision will be made as to continuation of the surveillance program.

6.0 DENSIFICATION POWER SPIKE PENALTY

VYNPC has requested a change in the technical specifications to remove a densification power spike penalty from the 8 x 8 fuel so that the operating limit on the maximum linear heat generation rate would be 13.4 kw/ft. In the cycle 3 analysis the fuel rod maximum local power was taken as 13.4 kw/ft and a .3 kw/ft power spike penalty was subtracted to give an operating limit of 13.1 kw/ft. For cycle 4, the maximum local power is assumed to be 13.7 kw/ft and the operating limit is 13.4 kw/ft. For applications where the bundle power, rather than the local power is the significant quantity (such as boiling transition calculations) the 13.4 value is used and the densification power spike factor (.3 kw/ft) is not applied.

GE described this method in Appendix B to reference (2) which we have not yet approved. Since the review is not complete, in the interim, Vermont Yankee must continue to use 13.4 kw/ft as the peak linear heat generation rate limit.

7.0 TECHNICAL SPECIFICATIONS

The proposed Technical Specification changes based on the reload submittal and GETAB incorporate the Fuel Cladding Integrity Safety Limit MCPR and Operating Limit MCPR's as identified in reference (1). The LTA is 8 x 8 fuel and the safety limit MCPR and operating MCPR for 8 x 8 fuel apply.

As discused in Section 6 above, VYNPC proposed to incorporate the effect of densification power spiking for 8 x 8 fuel into the maximum allowable LHGR without using a corrected equation. We have not yet approved this concept, thus VYNPS will be required to continue to use a correction equation to account for the effect of power spiking caused by fuel densification.

We find the proposed Technical Specification changes, with the exception of that identified above, acceptable and consistent with the information provided in the reload 3 licensing submittal.

8.0 LEAD TEST ASSEMBLIES

Based on our review of the information provided concerning the LTA's, the use of two LTA's in the VYNPS reload 3 cycle is acceptable. This acceptance does not allow expanded use of similar additional assemblies in VYNPS or in other reactors without further staff review of their specific application.

In order to facilitate future reviews in which fuel assemblies similar in design to the LTA's are used as a major portion of the reload, VYNPC is being requested to report the results of its findings concerning the LTA's after the four cycle irradiation is completed.

9.0 INSERVICE INSPECTION OF FEEDWATER BLEND RADII

By letter dated May 28, 1976, the NRC recommended to VYNPC a specific course of action to be taken with regard to the inspection of feedwater nozzle blend radii at the Vermont Yankee Nuclear Plant. By letter dated June 16, 1976, VYNPC acknowledged that the recommendations had been incorporated into the plans for the refueling outage.

Vermont Yankee performed an inservice inspection of the feedwater nozzles to determine if thermal fatique cracks were present in the nozzle inner blend radius. Initial procedures included removing the spargers and conducting a dye penetrant examination. This examination indicated numerous surface cracks. These cracks were ground down in increments of 1/16 inch until subsequent dye penetrant examination indicated that they had been removed. Prior to and following crack removal an ultrasonic examination using the recently developed, and still experimental, Breda technique was also conducted in an attempt to verify the presence of the cracks and their removal by grinding. The results of these volumetric examinations indicated possibly significant indications in the blend radius region. To better characterize these indications, additional ultrasonic examinations were conducted by the General Electric Company using techniques typically used by Vermont Yankee for scheduled Section XI inservice inspections. The results of this second examination showed no indications exceeding those from the clad-base-metal interface.

Based on our evaluation, we conclude that the margin of safety in the Vermont Yankee feedwater nozzles in their present condition, is not reduced below that level considered to be acceptable for continued operation for one more fuel cycle of 18 months or less.

We and our consultants (Sandia Laboratories) have independently reviewed the raw data from the Breda ultrasonic and GE tests at the site. Based on our review we conclude that the ultrasonic indications are not thermal fatique cracks.

An independent evaluation of the condition of the feedwater nozzles at Vermont Yankee, in so far as serviced-induced thermal fatique cracking was concerned, was made using fracture mechanics considerations and comparisons with reports from eight other BWR plants in terms of service life and the depths of cracks that were ground out. These other BWR plants had more severe cracking than Vermont Yankee, but the cracks were ground out before they affected the safety limits for operation.

In this evaluation, service life was used as the best indicator of the number occasions when low, intermittent flow of cold feedwater occurred.

Severity of cracking was measured in two ways: (a) the total depth of the deepest grindout, and (b) the sum of the depths of base metal penetration reported in all four feedwater nozzles. These quantities were used because only the low-cycle fatigue caused by intermittent feedwater flow would be expected to cause growth of cracks beyond 1/8 inch or so below the metal surface.

Based on the consideration given above, Vermont Yankee appears to have had less severe thermal cycling than some of the other BWR plants. The sum of base metal penetrations was less than that for four other plants, and the depth of the deepest grindout was 3/8 inch total (1/8 inch into the base metal), well below that at two other plants with comparable service life. Based on the trends in crack severity with service life, it can be said that additional service of 18 months or less should not produce crack depths that impinge upon the accepted margin of safety during the 18 month period.

In addition to our findings as stated above we recommend that:

1. At the next refueling shutdown, the feedwater nozzles should be reinspected. The method to be used and the extent of the inspection will be determined later, based on the final reports on the present inspection submitted by the licensee and on the results of development work on NDE inspection which are now in progress.

2. The licensee has agreed to submit a program within three months to reduce the incidence of intermittent flow of cold feedwater by changes in system procedure and/or equipment. We will review this material and notify the licensee of our conclusions.

Based on our review, independent evaluation, and our determination that an acceptable margin of safety exists, we conclude that operation of Vermont Yankee is acceptable.

10.0 ENVIRONMENTAL CONSIDERATIONS

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

11.0 CONCLUSION

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

Dated: August 2, 1976

REFERENCES

1. Reload Number 3 Licensing submittal, Vermont Yankee Nuclear Power Station, dated April 23, 1976, as supplemented May 25, 1976.

2. GE/BWR Generic Reload Licensing Application for 8 x 8 fuel, Revision 1, Supplement 3, September 1975, NEDO-20360.

3. GE/BWR Generic Reload Application for 8 x 8 fuel, Revision 3, NEDE-20360-IP, September 25, 1975.

4. V. A. Moore, letter to I.S. Mitchel, "Modified GE Model for Fuel Densification, Docket No. 50-321," March 22, 1974.

5. Linford, R. B., "Analytical Methods of Plant Transient Evaluation for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

6. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for GE/BWR Amendment No. 1," NEDO-10802-1, April 1975.

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

8. Letter from D. E. Vandenburgh to Office of Nuclear Reactor Regulation "Analytical Supporting Documentation for Operation of the Vermont Yankee Nuclear Power Station with Bypass Flow Holes Plugged," WVY-75-70, July 30, 1975.

9. "Safety Evaluation Report on the Reactor Modification to Eliminate Significant In-Core Vibration in Operating Reactors with 1-Inch Bypass Holes in the Core Support Plate," Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, February 1976.

10. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. DPR-28 issued to Vermont Yankee Nuclear Power Corporation which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station, located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications relating to the replacement of 136 of 368 fuel assemblies in the reactor core of VYNPS constituting refueling of the core for cycle 4 operation. Also, in addition to evaluating cycle 4 reload considerations, the Commission's related Safety Evaluation evaluates the inservice inspection of feedwater blend radii.

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

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

For further details with respect to this action, see (1) the application for amendment dated April 23, 1976, as supplemented May 25, 1976, (2) Amendment No. 25 to License No. DPR-28, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 2nd day of August 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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