September 30, 1977

Docket No. 50-271

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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. 39 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 applications dated March 9, 1977; June 8, 1977; July 1, 1977, as supplemented; and September 16, 1977; and staff discussions.

This amendment modifies the Technical Specifications relating to the replacement of 192 of 368 fuel assemblies in the reactor core of VYNPS constituting refueling of the core for cycle 5 operation.

In addition, this amendment: (1) raises from 10% to 20%, the power level below which the Rod Worth Minimizer must be operable, (2) incorporates into the Technical Specifications qualification requirements for the plant health physicist and the requirement that an individual qualified in radiation protection procedures be onsite when there is fuel in the reactor, and (3) changes the acceptance criterion for surface indications detected during the inservice inspection of Category F welds.

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

Sincerely,

Original Signed by

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

Enclosures and cc:
See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	OELD	C-ORB#4:DOR		
SURNAME	RIngram	JSiegel		RWReid		
DATE	9/ /77	9/ /77	9/30/77	9/30/77		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

Enclosures:
1. Amendment No. 39
2. Safety Evaluation
3. Notice

cc w/enclosures:

Mr. S. D. Karpyak
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, Esq.
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

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

Richard E. Ayres, Esq.
Natural Resources Defense Council
917 - 15th Street, N.W.
Washington, D.C. 20005

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

Anthony Z. Roisman, Esq.
Sheldon, Harmon & Roisman
1025 15th Street, N.W., 5th Floor
Washington, D.C. 20005

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

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

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

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection
Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D.C. 20460

Yankee Atomic Electric
  Company

U. S. Environmental Protection
  Agency
Region I Office
ATTN: EIS COORDINATOR
JFK Federal Building
Boston, Massachusetts 02203

cc w/enclosures and cy of VY's
  filing dtd.: 3/9/77, 6/8/77, 7/1/77, 9/16/77
Public Service Board
State of Vermont
120 State 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. 39
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 March 9, 1977; June 8, 1977; July 1, 1977, as supplemented; and September 16, 1977, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating 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 through Amendment No. 39 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
6 & 7	6 & 7
12	12
13 & 14	13 & 14
15-b	-
47	47
70R	70R
71 & 72	71 & 72
74	74
76 & 77	76 & 77
110b	110b
112 - 115	112 - 115a
125a	125a
180c	180c
-	180-01
190	190 & 190a

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

1.1 SAFETY LIMIT

C. 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 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 x 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:

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

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

D. 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.

1.1 (cont.)

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.

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.

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 at 105% of rated steam flow. 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 4.51% and 25.34%, 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 46.18% and 87.84% 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.

2.1 FUEL CLADDING INTEGRITY (Continued)

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.

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)
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)
1	b. Downscale (Note 7)			X	$\geq$2/125 full scale
1	Trip System Logic	X	X	X	
1	Scram Discharge Volume	X	X	X	$\leq$12 gallons

3.3 LIMITING CONDITIONS FOR OPERATION	4.3 SURVEILLANCE REQUIREMENTS
2. The control rod drive housing support system shall be in place when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:	3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:
(a) If after withdrawal of at least twelve control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or	(a) The Reactor Engineer shall verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.
(b) If all rods, except those that cannot be moved with control rod drive pressure, are fully inserted, no more than two rods may be moved.	(b) The Rod Worth Minimizer diagnostic test shall be performed.
	(c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciator of the selection errors verified.
	(d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.
4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that:	4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.
(a) When the reactor is critical and below 20% power the maximum calculated worth of any withdrawn increment of any in-sequence control rod which is not electrically disarmed shall be less than 1.3% delta k.	

3.3 LIMITING CONDITIONS FOR OPERATION	4.3 SURVEILLANCE REQUIREMENTS
(b) when the reactor is above 20% 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.	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. 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.	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 LIMITING CONDITIONS FOR OPERATION

C. Scram Insertion Times

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

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.358
36	25.34	1.096
26	46.18	1.860
06	87.84	3.419

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

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.379
36	25.34	1.164
26	46.18	1.971
06	87.84	3.624

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

4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

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

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

3.3 LIMITING CONDITIONS FOR OPERATION

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

3. Scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

F. If Specification 3.3A through D above are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.3 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

3.3 (Cont'd)

B. Control Rods

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

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

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

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

(1) NEDO-20360, Revision 1, Supplement 3, September 25, 1975.

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 20% power the consequence of a rod drop are less severe and the worths of rods in normal patterns are much less, therefore limiting rod worths to 2.0% delta k at power levels above 20% 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 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.

3.6 LIMITING CONDITIONS FOR OPERATION

J. <u>Thermal Hydraulic Stability</u>

When the reactor mode switch is in RUN, the reactor shall not intentionally be operated in a natural circulation mode, nor shall an idle recirculation pump be started with the reactor in a natural circulation mode.

4.6 SURVEILLANCE REQUIREMENTS

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.I.1, it shall be assumed that the facility had been on a 6 month inspection interval.

4. Once each refueling cycle, a representative sample of approximately 10% of the snubbers shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lbs need not be functionally tested.

Amendment No. ~~24~~, 39

TABLE 4.6.1

CATEGORY	EXAMINATION AREA	METHOD OF EXAMINATION	(5 year Period) EXTENT AND FREQUENCY OF EXAMINATION
REACTOR			
A	Longitudinal and circumferential shell welds in core region	-	Inaccessible due to existing vessel design
B	Longitudinal and circumferential welds in shell (other than those of Category A & C) and meridianal and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric	Closure Head: 3% of each meridianal weld, 1.5% of each circumferential weld Vessel and Bottom Head: Not accessible due to existing vessel design
C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	25% of each circumferential weld
D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	Volumetric	100% of nozzle-to-vessel weld and selected positions of inner radius sections of nozzle-to-vessel juncture; 25% of total nozzles subject to inspection
E-1	Control rod drive penetrations and control rod housing pressure boundary welds	Volumetric	Penetrations in this category meet the Exclusion Criteria of Section ISI-121
E-2	Control rod drive penetrations and control rod and control rod housing pressure boundary welds	Visual	10% of the total number of welds
F	Primary nozzles to safe-end welds	Volumetric & Visual or Surface	100% of the circumference of the safe-end weld; 25% of all safe-end welds. See Footnote #2
G-1	Pressure retaining bolting two inches and larger in diameter	Volumetric & Visual or Surface	25% of total number of bolts, studs, and nuts. Examination of subject headings, threads, and ligaments in base material of flanges shall be done only when the connection is disassembled for other reasons.

TABLE 4.6.1 (CONT'D)

CATEGORY	EXAMINATION AREA	METHOD OF EXAMINATION	(5 year Period) EXTENT AND FREQUENCY OF EXAMINATION
REACTORS			
G-2	Pressure retaining bolting below two inches in diameter	Visual	25% of total number of bolts, studs, and nuts, except for belting of a single connection meeting the exclusion criteria of In-Service Inspection Code Para. ISI-121
H	Integrity welded vessel supports	-	Not accessible due to existing vessel design
I	Closure head and vessel cladding	Head-Visual & Surface or Volumetric Vessel-Visual	Two patches in closure head, two patches in vessel, each patch to be 36 square inches
N	Interior surfaces and internal components of the reactor vessel	Visual	Those areas to examination which are made accessible by maintenance work and equipment removal during normal refueling outages
PIPING			
F	Vessel, pump and valve safe-ends to primary pipe welds and safe-ends in branch piping welds	Visual & Surface & Volumetric	100% of the circumference of each safe-end weld; 25% of all safe-end welds. See Footnote #2
G-1	Pressure retaining bolting two inches and larger in diameter	Volumetric & Visual	25% of total number of bolts, studs, and nuts while in place or when bolting is disassembled for other reasons
G-2	Pressure retaining bolting below two inches in diameter	Visual	25% of total number of bolts, studs, and nuts, except for bolting of a single connection meeting the exclusion criteria of In-Service Inspection Code paragraph ISI-121. Examinations to be performed in place or when bolting is disassembled for other reasons.

CATEGORY	EXAMINATION AREA	METHOD OF EXAMINATION	(5-year Period) EXTENT AND FREQUENCY OF EXAMINATION
PIPING			
J	Pressure containing welds in piping, longitudinal and circumferential seam welds	Volumetric & Visual	10% of the total number of circumferential joints, including one foot of all longitudinal welds from intersection with the selected circumferential weld joint
Note:	Whenever the system boundary is subjected to a hydrostatic test prior to plant startup subsequent to a refueling outage, the following criteria will be utilized:		
	Piping welds excluded from examination by ISI-121	Visual	25% of total number of welds. Insulation will not be removed.
K-1	Integrally-welded supports	Volumetric & Visual	10% of the total number of integrally welded supports within the system boundary
K-2	Piping supports and hangers	Visual	25% of all support members and structures
PUMPS			
L-1	Pump casing welds	Visual & Volumetric	See Footnote #1
L-2	Pump castings	Visual	See Footnote #1
F	Nozzle-to-safe-end welds	Volumetric & Visual	100% of the circumference of each safe-end weld 25% of all safe-end welds. See Footnote #2
G-1	Pressure retaining bolting two inches and larger in diameter	Volumetric & Visual	25% of total number of bolts, studs, and nuts while in place or when bolting is disassembled for other reasons.
G-2	Pressure retaining bolting below two inches in diameter	Visual	25% of total number of bolts, studs, nuts, except for bolting of a single connection meeting the exclusion criteria of In-Service Inspection Code paragraph ISI-120(d). Examination to be performed in place or when bolting is disassembled for other reasons.

TABLE 4.6.1 (CONT'D)

CATEGORY	EXAMINATION AREA	METHOD OF EXAMINATION	(5- year Period) EXTENT AND FREQUENCY OF EXAMINATION
PUMPS			
K-1	Integrally-welded supports	Volumetric & Visual	10% of the total number of integrally welded supports within the system boundary
K-2	Supports and hangers	Visual	25% of all support members and hangers
VALVES			
F	Valve-to-safe-end welds	Volumetric	100% of the circumference of each safe-end weld; 25% of all safe-end welds. See Footnote #2.
G-1	Pressure retaining bolting two inches and larger in diameter	Volumetric & Visual	25% of total number of bolts, studs, and nuts while in place or when bolting is disassembled for other reasons
G-2	Pressure retaining bolting below two inches in diameter	Visual	25% of total number of bolts, studs, and nuts, except for bolting of a single connection rect- ing the exclusion criteria of In-Service Inspec- tion Code paragraph ISI-121. Examinations to be performed in place or when bolting is disassembled for other reasons.
K-1	Integrally-welded supports	Volumetric & Visual	10% of the total number of integrally welded supports within the system boundary
K-2	Supports and hangers	Visual	25% of all support members and hangers
M-1	Valve body welds	Visual & Volumetric	See Footnote #1
M-2	Valve bodies	Visual	See Footnote #1

Footnote #1:

These categories fall at or near the end of the In-Service Inspection Internal (10 years). However, should the pumps or valves be dismantled during the 5 year program, the inspection may be performed at this time.

Footnote #2: See page 115a.

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Footnote #2

Category F welds, welded joints that are shown by surface examination to have indications* not exceeding (a), (b) or (c) are acceptable.

(a) For welded joints 5/8" or less in thickness, an indication with the maximum dimension $\leq$ 1/16".

(b) For welded joints greater than 5/8", but less than 2" in thickness, an indication with the maximum dimension $\leq$ 1/8".

(c) For welded joints greater than 2" in thickness, an indication with the maximum dimension $\leq$ 3/16'.

*Multiple aligned indications where the distance between adjacent indications is less than the length of the longer of these indications shall be combined when determining indication acceptability according to (a), (b), or (c).

3.6.I & 4.6.I SHOCK SUPPRESSORS (SNUBBERS)

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 4.6.2 as being in high radiation areas or especially difficult to remove need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impractability of testing such large units.

3.6.J THERMAL HYDRAULIC STABILITY

Not allowing operation in a natural circulation mode will provide additional stability margin, and it will provide protection against a reactivity insertion transient due to starting of an idle recirculation pump from the natural circulation mode.

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974
Subject: Hydraulic Shock Sway Arrestors

3.11 LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MCPR Operating Limit shall be equal or greater than the values shown on Figures 3.11-3A and -3B for 8x8 fuel and 7x7 fuel, respectively. 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.

Amendment No. ~~25~~, ~~34~~, 39

Figure 3.11-3A
MCPR Operating Limit for 8x8 Fuel

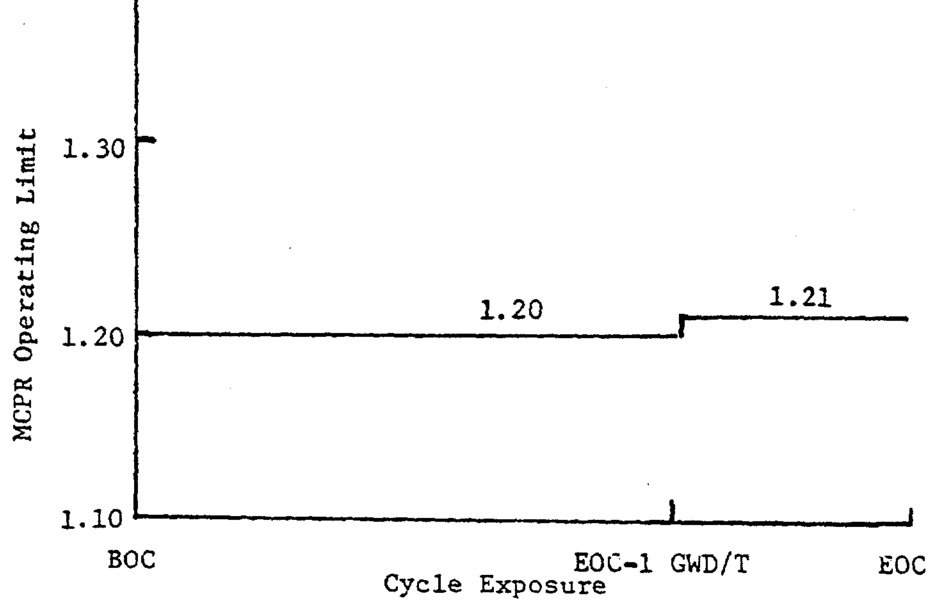
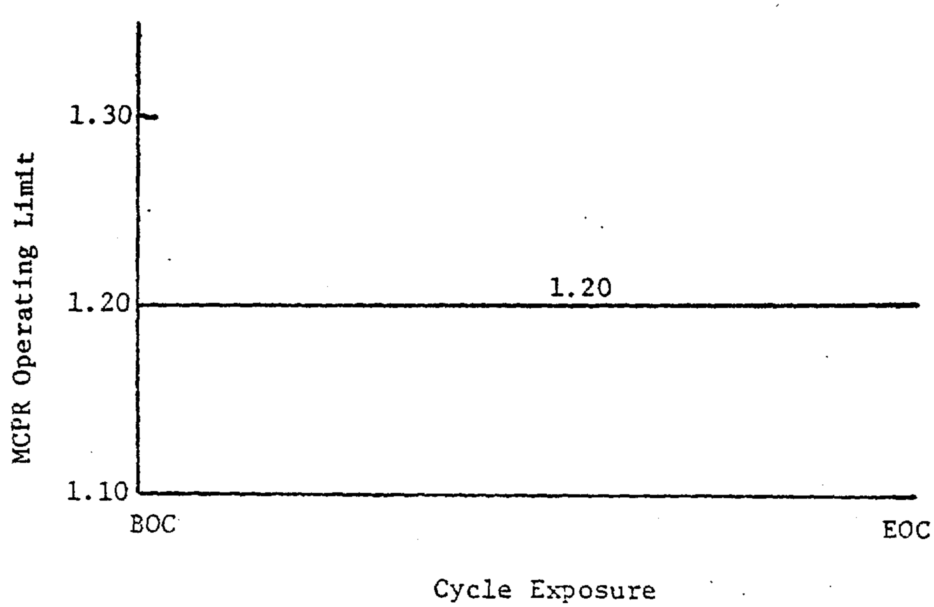


Figure 3.11-3B
MCPR Operating Limit for 7x7 Fuel



Amendment No. 39

6.0 ADMINISTRATIVE CONTROLS

Administrative controls are the written rules, orders, instructions, procedures, policies, practices, and the designation of authorities and responsibilities by the management to obtain assurance of safety and quality of operation and maintenance of a nuclear power reactor. These controls shall be adhered to.

6.1 ORGANIZATION

A. The Plant Superintendent or the Assistant Plant Superintendent has on-site responsibility for the safety and efficient operation of the facility.

B. The portion of the corporate management which relates to the operation of this plant is shown in Figure 6.1.1.

C. In all matters relating to the operation of the plant and to these Technical Specifications, the Plant Superintentent shall report to and be directly responsible to the Manager of Operations.

D. Conduct of operations of the plant is shown in Figure 6.1.2 and will be in accordance with the following minimum conditions (See Table 6.1.1).

1. A licensed Senior Operator and an individual qualified in radiation protection procedures* shall be present on site at all times when there is fuel in the reactor.

2. Licensed Operators on site shall be in accordance with Table 6.1.1, one of which must be in the control room at all times when fuel is in the reactor.

3. A licensed Senior Operator shall be in charge of any refueling operation.

4. Qualifications with regard to educational background experience, and technical specialities of the key supervisory personnel listed below shall apply and be maintained in accordance with the levels described in the American National Standards Institute N18.1-1971 "Selection and Training of Personnel for Nuclear Power Plants."

   a. Plant Superintendent
   b. Assistant Plant Superintendent
   c. Chemistry and Health Physics Supervisor
   d. Operations Supervisor
   e. Reactor and Computer Supervisor
   f. Maintenance Supervisor
   g. Instrument and Control Supervisor
   h. Shift Supervisors

*The requirement for an individual qualified in radiation protection procedures to be on site at all times when fuel is in the reactor will be implemented on 1/1/78.

5. The Plant Health Physicist shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 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 July 1, 1977[1], 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 the replacement of 192 fuel assemblies in the reactor core for cycle 5 operation. Eighty-four of the replacement fuel assemblies are fresh 8X8 fuel assemblies and the remaining 108 are 8X8 fuel assemblies which received one cycle of exposure in cycle 3. The average enrichment of the 192 assemblies to be loaded is 2.74 weight percent uranium-235. The 84 fresh fuel assemblies will have holes drilled in the lower tie plates. This addition of drilled fuel assemblies, in conjunction with the 136 drilled fuel assemblies which were loaded for cycle 4 will bring the core to about 60% of being a fully drilled core. Sixty of the fresh assemblies will have a high gadolinia poison content and the remaining 24 will have a low gadolinia content. These fuel assembly types are designated as 8D274H and 8D274L, respectively. Table 1 lists the fuel type and the number of assemblies in cycle 4 and cycle 5.

The neutronic, thermal-hydraulic, and mechanical designs of 8X8 fuel assemblies during normal operation, operational transients, and postulated accidents were evaluated, and found acceptable by the NRC staff in a Directorate of Licensing report entitled "Technical Report on the General Electric Company (GE) 8X8 Assembly" dated February 1974[3]. The use of 8X8 fuel assemblies for reloads was also reviewed, and found acceptable, by the Advisory Committee on Reactor Safeguards and discussed in its annual report dated February 12, 1974[4]. License Amendment No. 19, issued on March 11, 1976, specifically approved the use of 8X8 fuel assemblies at VYNPS. The technical support for the proposed reload is documented in both the specific documentation for the VYNPS as previously referenced and the Boiling Water Reactor reload licensing application for 8X8 fuel assemblies[5]. Although this report on 8X8 reloads is still under revision and NRC staff review, the report describes many safety analyses which were previously found acceptable and provides an acceptable analytical basis for the evaluation of the VYNPS reload with 8X8 fuel assemblies.

The documentation submitted in support of this reload application included a September 16, 1977, VYNPC letter which provided responses to our request for additional information dated September 1, 1977.

As directed by our Order dated March 11, 1977, VYNPC submitted a letter dated August 12, 1977, a reevaluation of the emergency core cooling system (ECCS) performance which was calculated in accordance with a revised, approved GE evaluation model[12]. This reevaluation corrected certain errors that existed in previous ECCS calculations and incorporated several modeling changes into the VYNPS ECCS analysis from the above GE model. We have reviewed the ECCS reevaluation of VYNPS to demonstrate that continued plant operation with the linear heat generation rates specified in our March 11, 1977 Order provides continued assurance that the ECCS will conform to the performance requirements of 10CFR50.46(b).

The proposed changes to VYNPS Technical Specifications which relate to the core reload include:

1. A change in the rod block monitor trip setting from $\leq 0.66W + 41\%$ to $\leq 0.66W + 40\%$ (where W is the full flow fraction).

2. A change from the percent insertion criterion for scram time surveillance requirements to a "drop-out-of-position" criterion.

3. A change to restrict plant operation in the natural circulation flow mode.

4. A change to preclude recirculation pump startup at reactor power operation from the natural circulation flow mode.

In separate and unrelated issues, VYNPS has requested the following changes to the VYNSP Technical Specifications:

1. By letter dated June 8, 1977, a change to the Technical Specifications which would require that the plant health physicist meet the qualification requirements of Regulatory Guide 1.8, September 1975 (Personnel Selection and Training) and that an individual qualified in radiation protection

procedures be onsite when there is fuel in the reactor. These proposed changes were in response to our letter dated April 5, 1977, on the same subjects.

2. By letter dated March 9, 1977, a change to the Technical Specifications which would increase from 10% to 20% the power level below which the Rod Worth Minimizer (RWM) must be operable.

3. By letter dated September 16, 1977, a change to the Technical Specifications which would change the acceptance criteria for indications detected during the inservice inspection of Category F welds.

Each of the above proposed changes to the Technical Specifications is discussed in this Safety Evaluation (SE).

TABLE 1

Fuel Type and Number

Fuel Type		Cycle 4	Cycle 5
Reload 1		40	40
Reload 2	8D219	192	108
Reload 3	8D274H	44	44
	8D274L	86	86
	LTA	2	2
	High Gd	4	4
Reload 4	8D274H	-	60
	8D274L	-	24
TOTAL		368	368

2.0 Evaluation

2.1 Nuclear Characteristics

The information presented in the VYNPC submittals follows the guidelines of NEDO-20360(5). The fuel assembly pattern consists of previous core and reload assemblies in a symmetrical pattern throughout the core. The low gadolinia reload assemblies are loaded in the exterior of the core and the high gadolinia reload assemblies are loaded in the interior regions of the core. The nuclear characteristics of the cycle 5 core are similar to that of the previous cycle. The total control system worth and the temperature behavior of the reconstituted core will not differ significantly from those values previously reported for the VYNPS.

The void coefficient of reactivity will range from $-12.17\text{X}10^{-4}$ to $-11.85\text{X}10^{-4}$ (% void)$.^{-1}$ The previous cycle values ranged from $-11.33\text{X}10^{-4}$ to $-10.33\text{X}10^{-4}$ (% void)$.^{-1}$ The fuel temperature coefficient at 650°C changes from a cycle 4 range of $-1.226\text{X}10^{-5}$ through $-1.358\text{X}10^{-5}$°F^{-1} to a range of $-1.138\text{X}10^{-5}$ through $-1.234\text{X}10^{-5}$°F for cycle 5.

The shutdown margin meets the Technical Specification requirements that the core be at least 0.25%Δk subcritical in its most reactive state with the most reactive rod fully withdrawn and all others fully inserted. For cycle 5 a minimum shutdown margin of 1.01%Δk was calculated. Reference 1 calculational results indicate that a boron concentration of 800 ppm in the moderator will bring the reactor subcritical by 6.4%Δk at 20°C and under a xenon free condition. Therefore, the alternate shutdown requirement of the General Design Criteria 26 of Appendix A to 10 CFR Part 50 is met by the Standby Liquid Control System which contains the borated solution.

VYNPC has proposed changes to the Technical Specifications to convert from the "% insertion" criterion for scram time to a "drop-out-of-position" criterion. This change was proposed simply to provide a more convenient method for checking scram time against the Technical Specification requirements. Currently, the Technical Specifications prescribe that control rods shall be 50% inserted within a specific average time. Since the % insertion values used in the specifications fall between specific control rod notch positions, and only control rod notch positions are indicated in the test data, interpolation has to be done in order to determine if the test data are within the specifications. Specifying scram time on the basis of notch position switch indication would remove the necessity for interpolating the data, thus reducing the chance of errors. These changes have been reviewed, and we conclude that the revised method is entirely consistent with the requirements of the current approved scram time Technical Specification and, therefore, is acceptable.

Thus, based on the information presented in the VYNPC submittals and supplemented by the applicable sections of the generic 8X8 reload report, the nuclear characteristics and performance of the reconstituted core for cycle 5 operation are acceptable.

2.2 Mechnical Design

The reload 4 fuel assemblies have the same mechnical configuration and fuel assembly enrichments as the 8D274L and 8D274H fuel assemblies described in the 8X8 generic reload report. The 8D219 reload 2 fuel is identical to the current 8D219 fuel in the VYNPS core. On the bases of the VYNPC submittals, our safety evaluation on lower tie plate drilling (incore vibrations)[6], and the substantial operating experience of these fuel types in operating reactors[1], we conclude that the mechanical design of the fuel proposed for cycle 5 operation at the VYNPS is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8X8 fuel reload topical report[15] and GETAB[7] Licensing Topical Report are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins for cycle 5. Application of thermal-hydraulic analyses involves:

1. the determination of the fuel damage safety limit Minimum Critical Power Ratio (MCPR); and
2. the operating limit MCPR such that, for any anticipated transient, the safety limit MCPR will not be violated.

We have evaluated the cycle 5 thermal margins based on the GETAB report and plant specific input information provided by VYNPC. Our evaluation is summarized herein.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during normal operation or transients that are anticipated to occur with moderate frequency. The uncertainties in the core and system operating parameters and the General Electric Critical Length (GEXL) correlation[7] form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for cycle 5 are the same as those used in the reference cycle.

We have determined that the thermal hydraulics performance has been conservatively considered for cycle 5 operation for VYNPS. The calculation methods and input data have been conservatively represented and used in this analysis. Therefore, we find the safety limit MCPR of 1.06 to be acceptable for VYNPS cycle 5 operation.

2.3.2 Operating Limit MCPR

To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in MCPR (ΔCritical Power Ratio (CPR)). VYNPC has submitted the results of analyses of those transients[1]. The addition of these ΔCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR required to avoid violation of the safety limit, should the limiting transient occur. Transients analyzed included the turbine trip without bypass, a loss of feedwater heater, and a rod withdrawal error.

The most limiting transient was the turbine trip without bypass from 1 GWD/T before end-of-cycle (EOC) to EOC. The ΔCPR for this transient was 0.15. Prior to 1 GWD/T before EOC, the turbine trip without bypass was analyzed at various times during the cycle using the appropriate scram reactivity curves (the scram reactivity curve decreases with increasing burnup). The ΔCPR generally increases with burnup to EOC for this transient. The rod withdrawal transient analysis is performed at beginning-of-cycle (BOC) to establish the rod block monitor set point. The ΔCPR for this transient was calculated to be 0.14. VYNPC has proposed to change the rod block monitor trip setting from ≤ 0.66 W + 41% to ≤ 0.66 W + 40% (W equals the full flow fraction) to make the Technical Specifications consistent with the analysis.

We have reviewed the calculational methods referenced by VYNPC, the input data, and safety analysis justification for calculations of operational transients. The operating limit MCPR's as proposed in the Technical Specification changes for cycle 5 operation are acceptable.

2.4 Accident and Transient Analysis

2.4.1 Operational Transients

VYNPC 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 transients parameter deviations were reanalyzed"[1]. The methods of these analyses are described in Reference 5. The input parameters and functions which characterize this cycle are analyzed under EOC conditions, and also at an intermediate exposure as previously discussed. The results of the analyses are given in Section 6 of Attachment 2 to Reference 1. The highest ΔCPR occurs for the 8X8 fuel during the turbine trip without bypass transient. The value obtained was 0.15 at the EOC 5. This corresponds to the 1.21 (1.06+ 0.15) EOC 5 8X8 fuel MCPR value of the preceding section. A brief description of the transients that were reanalyzed and reported in Reference 1 is presented in the following sections.

2.4.1.1 Overpressure Analysis

Reference 1 presents the results of an overpressure analysis to demonstrate that an adequate margin exists below the American Society of Mechanical Engineers Pressure Vessel Code allowable vessel pressure of 110% of vessel design pressure for VYNPS. This analysis demonstrates the adequacy of the safety/relief valves which are to be in service for cycle 5 operation. The transient analyzed was the closure of all main steam isolation valves (MSIV's) with high neutron flux scram. The analysis was performed for 105% power, scram initiated by high neutron flux, void reactivity conservatively applicable to this reload and credit for the relief function of the safety/relief valves with all safety valves operative. With these assumptions, a peak pressure of 1286 psig at EOC was calculated. Generic analysis applied to VYNPS showed that for this overpressure event, the failure of one safety valve would cause the maximum vessel pressure to increase by 20 psig. The effect of initiation of the transient from the high pressure trip setpoint rather than the value assumed in the analysis has been shown to have <20 psig effect on the transient result.[3] Hence, the maximum peak pressure at the vessel bottom for MSIV closure with an indirect scram, and one failed safety valve is calculated to be <1326 psig. This valve is about 50 psig below the Pressure Vessel Code allowable of 1375 psig, and is acceptable.

2.4.1.2 Rod Withdrawal Error

VYNPC has analyzed the Rod Withdrawal Error transient according to the assumptions given in Reference 1. The results show ΔCPR's of 0.14. The rod block monitor (RBM) setpoint will stop rod withdrawal at a MCPR of >1.06, the MCPR safety limit. Based on this analysis of worst case conditions for VYNPS, the proposed RBM flow biased setpoint relationship is acceptable for cycle 5 operation.

2.4.1.3 Turbine Trip with Bypass Failure

Fast closure of the turbine stop valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine stop valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator rotor. This closing, concurrent with the failure of the bypass valve system, causes a sudden reduction in steam flow which results in a nuclear system pressure increase and shutdown of the reactor. Again, this is the most restrictive ΔCPR transient for MCPR.

The VYNPC has performed this analysis with the safety/relief valve configuration for cycle 5 operation at VYNPS. The results have been discussed previously and are acceptable.

2.4.1.4 Loss of 100°F Feedwater Heater

The loss of a feedwater heater is the most limiting cool water injection transient. A feedwater heater can be lost by (1) the steam extraction line to the heater being closed off which removes the heat supply to the heater and causes a gradual cooling down of the tubes or (2) the feedwater flow through the heater being switched to the bypass line. In either case, the reactor will receive cooler feedwater which results in an increase in core inlet subcooling, and an increase in core power due to a negative dynamic void coefficient. The results of this transient, documented in Table 6-1 to Attachment 2, Reference 1, are not limiting and are acceptable.

2.4.2 Accident Analysis

2.4.2.1 ECCS Appendix K Analysis

In December of 1976, the NRC staff was informed that certain input errors and computer code errors had been made in the VYNPS ECCS analysis that was provided under the requirements of 10CFR50.46. An Order was issued to the Vermont Yankee Nuclear Power Corporation on March 11, 1977[10], requiring that corrected "revised calculations fully conforming to the requirements of 10CFR50.46 are to be provided for the (Vermont Yankee) facility as soon as possible. Our Order allowed VYNPS to continue operating with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values then existing in the Technical Specifications because the NRC staff was aware that revised modeling changes in the General Electric Company (GE) ECCS evaluation model would offset the effect of the errors. On April 12, 1977[12], the NRC staff approved the revised GE ECCS calculational model which incorporated the model changes.

As directed by our Order, VYNPC submitted corrected analyses for the present reload[2]. The analyses included correction of all input errors previously made and correction of all computer code errors. The analyses were performed using the calculational model which contains several model changes approved by the NRC staff in its Safety Evaluation issued April 12, 1977[12]. Our Safety Evaluation[12] is applicable to VYNPS and is incorporated by reference hearin.

The analyses submitted in Reference 2 were performed for the proposed cycle 5 fuel loading (Table 1). The cycle 4 and cycle 5 fuel loadings are comparable except that the cycle 5 core will be 60% drilled whereas the cycle 4 core was 37% drilled. Drilled fuel represents a major pathway for core spray to reach the lower plenum following a LOCA thereby providing earlier reflooding and lower calculated peak cladding temperatures. From an ECCS perspective the additional drilled fuel, which improves the ECCS effectiveness, is the only significant difference between the cycle 5 core and the cycle 4 core which was present when we issued our March 11, 1977 Order to VYNPC.

The VYNPS ECCS analysis incorporates by reference the FitzPatrick Nuclear Power Plant (FitzPatrick) ECCS analysis[9] which included a correction of the input errors and incorporated the revised GE ECCS model changes. FitzPatrick is similar to VYNPS in that both plants are BWR/4 reactors with the low pressure coolant injection (LPCI) system modification (our FitzPatrick Safety Evaluation[9] discusses in detail the nature of the LPCI modification).

The FitzPatrick ECCS analysis represents a lead plant analysis which provides the basis for the selection of the most limiting break location and single failure which yields the highest peak cladding temperature (PCT). The lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. This Safety Evaluation contains the results of our review of the non-lead plant analysis for VYNPS.

The analyses submitted in Reference 2 represent the first non-lead plant analysis (that references FitzPatrick as the lead plant) to be submitted with the corrected model. The analyses provide all information requested in our letter to GE on June 30, 1977, on the number of breaks to be analyzed, documentation to be provided, etc. for the new analyses. Since these analyses reference FitzPatrick as the lead plant analyses for BWR/4 plants with the low-pressure-coolant-injection (LPCI) system modification, the following description is provided of particular features of the FitzPatrick analyses as compared to the VYNPS analyses.

The FitzPatrick break spectrum (i.e., peak cladding temperature (PCT) versus break size) shows that the particular break producing the highest PCT for FitzPatrick is a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break. The particular reasons why that size and location break is limiting for FitzPatrick are stated in detail in the FitzPatrick Reload SE(9). For the same reasons that are stated in the FitzPatrick Reload SE, the limiting break location for VYNPS is the recirculation discharge line.

However, unlike FitzPatrick where the worst size break at that location was 80% of the maximum pipe area, for VYNPS the worst size break at that location is the design basis accident (DBA), or 100% of the maximum pipe area. As explained in the FitzPatrick SE:

1. FitzPatrick has a relatively small peripheral bypass area due to a relatively large number of fuel bundles in a relatively small reactor vessel. This makes FitzPatrick more likely than other plants to experience counter-current-flooding (CCFL) effects in the bypass region.

2. FitzPatrick has holes drilled in the lower tie plates of all fuel bundles to enhance flow in the bypass area. These holes, at the bottom of the bypass region, are a major pathway for core spray water to reach the lower plenum following a LOCA and thereby contribute to the reflooding inventory, providing earlier reflooding and lower calculated PCT's. Any CCFL effects in the bypass area will delay such reflooding, causing a higher calculated PCT.

The FitzPatrick Reload SE explains how the above two effects combine with the effect of slower depressurization for smaller breaks (i.e., more lower plenum flashing steam is produced later for smaller breaks, which is when spray water is trying to go down through the bypass region). These effects combine to make the 0.8* DBA break limiting for FitzPatrick.

VYNPS is much less sensitive to steam CCFL effects in the bypass region than is FitzPatrick. VYNPS has a larger bypass region which makes it less sensitive to CCFL effects. Also, VYNPS has only 220 of a total of 368 fuel bundles drilled (60%), whereas FitzPatrick has all 560 drilled (100%). Therefore, a smaller fraction of the spray water goes through the bypass region in VYNPS than in FitzPatrick due to the lesser number of holes, and the water that does go through the region is less affected by steam due to the larger area present at the top of the peripheral bypass area, where CCFL effects occur.

Therefore, the effects of more steam being produced later for smaller breaks do not dominate in VYNPS to produce a smaller-than-maximum size limiting break. Instead, the predominate effects for VYNPS are earlier departure from nucleate boiling and earlier high power plane uncovery for large break analyses compared to smaller break analyses. These effects cause the largest size discharge line break to be limiting for VYNPS.

For the above reasons, we concur with the analyses provided by VYNPC that the limiting break for VYNPS is the largest recirculation discharge line break. The MAPLHGR Technical Specifications earlier referenced are consistent with the analyses of that break.

We have reviewed the corrected analyses submitted for the reload in Reference 2. We conclude that VYNPS will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Reference 2, which are higher than the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values specified in our Order(10), and 2) when it is operated at a MCPR equal to or greater than 1.20 (MCPR limits which meet this criterion are currently required for reasons not connected with the LOCA as described elsewhere in our evaluation). Therefore, our findings of the Order of March 11, 1977, are still applicable and appropriate for the VYNPS, and continued operation under the lower MAPLHGR values setforth in the Order is acceptable for cycle 5. Final action to terminate the Order is expected shortly. However, until such action is taken, the conditions of the Order remain in effect.

2.4.2.2 Main Steam Line Break, Refueling Accident

The analyses of the following accidents were listed by VYNPC as being covered by the Final Safety Analysis Report (FSAR).

Main Steam Line Break Accident

Refueling Accident

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

2.4.2.3 Control Rod Drop Accident

The technical bases (bounding analyses) which are presented in Reference 5 were used to verify that the results of a rod drop excursion in the reloaded core would not exceed the design criteria.

For application to VYNPS reload 4, the actual Doppler coefficient, accident reactivity shape functions and scram reactivity functions are compared with the bounding analysis based in Figures 7.1 through 7.5 of Reference 1. Since the maximum values of the parameters after this reload will be well below the boundary value, the consequences of a rod drop transient from any insequence control rod would be below the 280 cal/gm design limit.

2.4.2.4 Fuel Loading Error Event

The fuel loading error event assumes that either 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. For VYNPS the case of the fuel bundle inserted in an improper location gave the most restrictive results.

Analysis of the load error accident results in the following peak linear heat generation rates (LHGR) and MCPR's in the misplaced bundle.

	MCPR	Peak LHGR
Mislocated 8D219	0.84	19.0 kw/ft
Mislocated 8D274	0.95	16.8 kw/ft
Rotated 8D274 (Reload 4)	0.99	16.5 kw/ft

GE has concluded that fuel bundles adjacent to the misplaced bundle are insignificantly affected by the presence of the misplaced bundle.

Since this accident results in a CPR less than the safety limit, it is expected that some of the fuel rods in the bundle will experience boiling transition. Detection of any abnormal fuel degradation is accomplished 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 radioiodine is required by Technical Specifications if a change in the off-gas activity of 25% or 5000 uCi/sec (whichever is greater) is detected.

The allowable limit for iodine in the reactor coolant, 1.1 uCi/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 setpoints 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 VYNPS 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 was 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 Quality 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 reviewed the procedures and video techniques for verifying that the fuel is correctly placed in the core. The inspection was completed September 26, 1977.

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 analysis for a Fuel Loading Error Accident at VYNPS to be acceptable.

2.4.2.5 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in Reference 1. The results of the cycle 5 analysis show that the 7X7 and 8X8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is well within the operational design guide in terms of decay ratio. Calculations were also performed by VYNPC to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

We have expressed generic concerns regarding the least stable reactor conditions allowed by Technical Specifications. These conditions could be reached by operation in the natural circulation mode. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. Our concerns relate to both the consequences of operating at an ultimate decay ratio and the ability of analytical methods to accurately predict decay ratios. GE is addressing our concerns through meetings, topical reports and a test program.

A license amendment for Peach Bottom Unit No. 2 was approved that authorized a reactor core stability test program which was performed at the end of cycle 2. The proposed test program is aiding in the resolution of our generic concerns on stability. The testing was performed during April 1977. The preliminary results from the testing have been provided to us by GE. The results show that the calculational techniques for evaluation of reactor stability margins are generally conservative.

Until this issue has been resolved generically, we have imposed a requirement on VYNPS which will restrict planned operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during cycle 5. On the basis of the foregoing, we consider the thermal-hydraulic stability of VYNPS to be acceptable.

3.0 Recirculation Pump Startup From the Natural Circulation Operational Mode

During a recent Boiling Water Reactor reload review, the question of recirculation pump startup from the natural circulation operational mode was raised. This pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence had not been previously evaluated, so that for this reload review, additional information was requested. VYNPC was requested to provide analyses and startup test results, to show that the startup of recirculation pumps from natural circulation conditions does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed. As discussed in Section 2.4.2.5 VYNPC agreed to incorporate Technical Specifications which preclude natural circulation mode operation thereby eliminating any possibility of a recirculation pump startup from the natural circulation mode. We find this measure to be acceptable.

4.0 Physics Startup Testing

As part of our review of cycle 5, we provided VYNPC with a description of the Physics Startup Testing program. Additionally, the program was discussed with VYNPC for clarification of reporting requirements and scheduling. VYNPC agreed to submit the results of the physics startup test program to the NRC staff 45 days after the completion of the startup program. We find the Startup Physics Testing program and reporting schedule are acceptable.

5.0 Technical Specifications

We have reviewed the proposed Technical Specification changes related to the reload submittal as identified in references (1), (2) and (8). In addition to minor editorial changes, the proposed changes involve a change in the rod block monitor trip setting, a change from the percent insertion criterion for scram time surveillance requirements to a "drop-out-of-position" criterion, a restriction on plant operation in the natural recirculation flow mode, a change to preclude recirculation pump startup at reactor power operation from the natural recirculation flow mode, and the addition of new exposure dependent MCPR operating limits. All changes have been discussed and evaluated above. We have determined that the proposed changes are either consistent with the analyses presented by VYNPC in support of cycle 5 operation, or are the result of NRC staff concerns relating to reactor core stability, and are therefore, acceptable.

6.0 Rod Worth Minimizer (RWM)

In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. VYNPS Technical Specifications 3.3.B.3 and 4 require the RWM to be operable below 10% of rated thermal power. Use of the RWM prevents the selection of high reactivity worth control rods which if upon withdrawal were to become uncoupled and subsequently dropped out of the reactor core, could cause fuel damage. The bases for requiring RWM operability up to 10% of rated thermal power are analyses [13] which indicate that above 10% power, even single operator errors cannot result in a dropped rod accident which could cause fuel damage.

More recent analyses [14] by GE indicated that the RWM is assumed to be operable below 20% of rated thermal power to prevent fuel damage as a result of the postulated dropped rod accident. Upon being informed by GE of the inconsistency between the most recent safety analysis [14] and the Technical Specifications, VYNPC adopted an administrative procedure to utilize the RWM below 20% power and requested a license amendment to change Technical Specifications 3.3.B.3 and 4.

The application for license amendment dated March 9, 1977, proposed a requirement for RWM operability below 20% of rated thermal power as compared with the existing requirement of 10% power. The proposed change is necessary to conform the Technical Specifications to assumptions used in rod drop analyses for VYNPC's cycle 4 application which we approved on August 2, 1976 and the present cycle 5 application. The change requiring RWM operability at up to 20% power level increases a safety margin and we find this acceptable.

7.0 Plant Health Physicist Qualification Requirements

VYNPC's proposed changes, submitted by letter dated June 8, 1977, would require that the plant health physicist meet the minimum qualification requirements of Regulatory Guide 1.8, Revision 1, 1975, and that an individual qualified in radiation protection procedures be onsite when there is fuel in the reactor (the latter requirement is to be implemented on January 1, 1978). Regulatory Guide 1.8 states that the plant health physicist should have a bachelor's degree or the equivalent in a science or engineering subject including some formal training in radiation protection. We have established guidelines to clarify the intent of "or the equivalent." Our definition of "equivalent" is as follows: Equivalent, as used in Regulatory Guide 1.8 for the bachelor's degree requirement, may be met with any one of the following: (a) 4 years of formal schooling in science or engineering, (b) 4 years of applied radiation protection experience at a nuclear facility, (c) 4 years of operational or technical experience/training in nuclear power, (d) any combination of the above totaling 4 years. It should be noted that the above requirement is in addition to the requirement for professional experience in applied radiation protection as specified in the Guide. We have modified the changes proposed by VYNPC and they have concurred. The revised Technical Specifications agree with the NRC staff positions, are consistent with those of other operating facilities, and are acceptable.

8.0 Inservice Inspection of the Reactor Coolant Pressure Boundary

8.1 Feedwater Nozzles and Control Rod Drive Return Line Nozzle

By letter dated July 19, 1977, we requested VYNPC to advise us of action to be taken during the present outage with regard to the inspection of the control rod drive (CRD) return line nozzle and the feedwater nozzle blend radii. By letter dated August 15, 1977, VYNPC advised us that they would inspect both of the above areas for cracks using a dye penetrant examination.

An inservice inspection of the accessible areas (without removing spargers) of the feedwater nozzles at VYNPS during the current outage revealed no indication of cracking. As a result, VYNPC intends to schedule the next inservice examination of the feedwater nozzles in about 2 years or at the next scheduled refueling outage after 20 but prior to 40 startup/shutdown cycles following the current refueling outage. This is in accordance with the latest NRC staff guidance as set forth in NUREG 0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking", July 1977.

VYNPC also examined the CRD return line nozzle during the current refueling outage. Using a liquid penetrant test, VYNPC detected 19 indications in the stainless steel cladding of the nozzle. The thermal sleeve was removed for this inspection. VYNPC classified the indications as surface porosities, rounded indications, and small cracks. The maximum crack length and depth were about 0.25 inches and 0.188 inches respectively. The surface irregularities were ground down in increments of 1/16 of an inch until penetrant examination revealed that all irregularities had been removed.

VYNPC notified us that they had prepared a contingency plan to reroute the CRD return line to the reactor water cleanup system per the recommendations in the General Electric Company's "BWR Services Information Letter SIL 200, Supplement 1" referenced in our July 19, 1977 letter, in the event that the cracks approached the critical flaw size of the reactor vessel (approximately 0.5 inches). The penetrant examinations indicated this was not the case, and VYNPC chose only to remove the surface indications, and install a new thermal sleeve of the type proviously installed (double concentric sleeve with flow shroud bolted to wall in 3 locations). VYNPC's next inservice inspection of the CRD return line nozzle is scheduled for the second refueling outage which follows the current outage.

We reviewed the actions taken by VYNPC in removing the irregularities and cracks detected in the CRD return line nozzle, their determination not to reroute the CRD return line nozzle, and their planned frequency of inservice inspection of this nozzle which they have scheduled about two years hence.

We have requested that, in accordance with NUREG 0312, VYNPC contact us at least 90 days prior to their next refueling outage at which time we can be appraised of their operating history to date. At this time, we can also inform them of any pertinent information, obtained from other operating facilities during the interim period, which may necessitate earlier inspection of feedwater nozzles or the CRD return line nozzle at VYNPS. Based on our ongoing review of this matter and the results of the completed nozzle examinations discussed above, we have determined that continued use of the feedwater nozzles and CRD return line nozzle as designed at VYNPS is acceptable during the next 2 fuel cycles.

8.2 Recirculation System Suction Nozzles

By letter dated September 16, 1977 (proposed Change No. 65) VYNPC requested that the VYNPC Technical Specifications be amended to permit use of inservice examination acceptance standards of IWB-3514.4 of the 1977 Edition of Section XI of the Pressure Vessel Code for

Category F (dissimilar metal) welds. The request resulted from an inservice examination of the recirculation line suction nozzles during the current outage which revealed several liquid penetrant indications in the N1B nozzle to safe end weld. VYNPC stated that the indications were approximately 3/32 inches in length and were typical of weld-type indications which may remain after welding and may become apparent during subsequent liquid penetrant examinations. The indications generally involved either slight slag inclusions, slag lines, or narrow tight lines of oxides. In addition, some fine porosity-type "inclusions" were apparent. With regard to future inservice inspection, we recommend that at the next refueling outage, the recirculation line suction nozzles be inspected with an appropriate penetrant examination.

The basis of the VYNPC request is that the acceptance standards currently required by the VYNPS Technical Specifications are inconsistent with those permitted by the 1977 Edition of Section XI for inservice examinations. Specifically, the VYNPS Technical Specifications refer to Section III of the Pressure Vessel Code which does not permit any linear indications, whereas IWB-3514.4 permits linear indications from 0.2 to 0.8 inch length, depending on material and thickness.

We have reviewed the acceptance standards in the 1977 Code Edition and have concluded that these standards which permit linear indications of 0.2 to 0.8 inch lengths, depending on material and thickness, could be nonconservative. These standards are deduced from inservice volumetric examination acceptance standards for surface indications by assuming a semi-circular surface flaw. Although some of these inservice volumetric examination surface acceptance standards are developed from fracture mechanics considerations, the standards do not evaluate the relevancy of these surface indications, e.g. cracks, machining marks or other surface conditions, etc. Since surface examination alone will not measure the depth of a surface indication, which is the critical parameter in estimating the structural integrity of a flawed weld, the proposed surface acceptance standards were evaluated to be excessive and nonconservative.

In consideration of the above concerns, we have concluded that the allowance of surface indications more conservative than those contained in IWB-3514.4 provides for the maintenance of structural integrity for Category F welds. Thus, we have recommended the following as an alternative to the VYNPC proposal:

For Category F welds, welded joints that are shown by surface examination to have indications* not exceeding (a), (b) or (c) are acceptable.

(a) For welded joints 5/8" or less in thickness, an indication with the maximum dimension $\leq$ 1/16".

(b) For welded joints greater than 5/8", but less than 2" in thickness, an indication with the maximum dimension $\leq$ 1/8".

(c) For welded joints greater than 2" in thickness, an indication with the maximum dimension $\leq$ 3/16".

*Multiple aligned indications where the distance between adjacent indications is less than the length of the longer of these indications shall be combined when determining indication acceptability according to (a), (b) or (c).

We have discussed this modification of the VYNPC proposed Technical Specification and they have concurred.

9.0 Environmental Consideration

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

10.0 Conclusion

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

Dated: September 30, 1977

REFERENCES

1. Letter, D. E. Vandenburgh to NRC, WVY 77-62, dated July 1, 1977.

2. Letter, W. P. Johnson to NRC, WVY 77-71, dated August 12, 1977.

3. "Technical Report on the General Electric Company 8X8 Assembly," NRC Directorate of Licensing, February 1974.

4. Annual Report by Advisory Committee on Reactor Safeguards, February 12, 1974.

5. GE/BWR Generic Reload Licensing Application for 8X8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1976.

6. "Safety Evaluation Report on the Reactor Modification to Eliminate Significant Incore Vibration in Operating Reactors with 1-inch Bypass Holes in Core Support Plates," NRR, NRC, February 1976.

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

8. Letter, D. E. Vandenburgh to NRC, dated September 16, 1977.

9. Letter, R. W. Reid (NRC) to G. T. Berry (Power Authority of the State of New York), September 16, 1977.

10. Letter, R. W. Reid (NRC) to D. E. Vandenburgh (VY), dated March 11, 1977.

11. Letter, R. W. Reid (NRC) to R. Groce (VY), November 12, 1975.

12. K. R. Goller (NRC) to G. C. Sherwood (GE), dated April 12, 1977.

13. "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972, Supplement 1, July 1972, and Supplement 2, January 1973.

14. "GE/BWR Generic Reload Licensing Application for 8X8 Fuel," Revision 1, Supplement 3 (NEDO-20360), September 25, 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 U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation (the licensee), which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (VYNPS) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

This amendment modifies the Technical Specifications relating to the replacement of 192 of 368 fuel assemblies in the reactor core of VYNPS constituting refueling of the core for cycle 5 operation.

In addition, this amendment: (1) raises from 10% to 20%, the power level below which the Rod Worth Minimizer must be operable, (2) incorporates into the Technical Specifications qualification requirements for the plant health physicist and the requirement than an individual qualified in radiation protection procedures be onsite when there is fuel in the reactor, and (3) changes the acceptance criterion for surface indications detected during the inservice inspection of Category F welds.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations 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, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 3, 1977; June 8, 1977; July 1, 1977, as supplemented; and September 16, 1977, (2) Amendment No. 39 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 30th day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile

Morton B. Fairtile, Acting Chief
Operating Reactors Branch #4
Division of Operating Reactors