JAN 28 1974

Docket No. 50-271

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Albert A. Cree, President
77 Grove Street
Rutland, Vermont 05701

Change No. 15
License No. DPR-28

Gentlemen:

As discussed in our letter dated January 17, 1974, authorizing Change No. 13 to the Technical Specifications of Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station, we are continuing our review of the entire Technical Specifications and the reissuance of the Technical Specifications on a section by section basis.

On the basis of our review reflected in the enclosed Safety Evaluation, we have concluded that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed by replacing Sections 3.5, 3.6, 3.7, 4.5, 4.6, and 4.7 (pages 84 through 140) in their entirety with the enclosed sections.

As discussed between our respective staffs and stated in our related Safety Evaluation, Vermont Yankee is planning modifications to the following systems prior to or during the next refueling outage. Some of these modifications will result in a need for additional changes to these approved Technical Specification changes as discussed in the enclosed Safety Evaluation. The affected systems include: (1) Automatic Depressurization System (Specification 3.5.F), (2) Pressure Suppression Chamber-Drywell Vacuum Breakers (Specifications 3.7.A.4 and 4.7.A.4), and (3) Containment Air Dilution System.

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During the interim period between the issuance of this Technical Specification change and the operation of the drywell vacuum breaker surveillance system (Specification 4.7.A.4), Vermont Yankee shall perform daily checks on the drywell vacuum breakers to assure that the valves are closed and shall perform weekly the drywell vacuum breaker tests specified in Specification 4.7.A.4.a., except those related to the position switches, indicators, and alarms. As stated in your letter of December 20, 1973, installation and operation of this system is scheduled for no later than June 1, 1974.

As discussed in the enclosed Safety Evaluation, you are requested to submit your schedule, within 30 days of the date of this letter, for installation and operation of a redundant conductivity monitoring system in the reactor coolant system. We understand that purchase delays for components of this system may not allow installation during the next refueling outage.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Technical Specifications (pages 85 through 146)

cc w/enclosures: See next page

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OFFICE ➤ X7403	L:ORB #2	L:ORB #2	L:ORB #2	L:OR		
SURNAME ➤	FDAnderson:sjh	RMDiggs	DLZiemann	DJSkovholt		
DATE ➤	1/23/74	1/23/74	1/25/74	1/28/74		

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

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

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

Richard E. Ayres, Esquire
David Schoenbrod, Esquire
National Resources Defense Council, Inc.
15 West 44th Street
New York, New York 10036

Honorable Kimberly B. Cheney
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

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

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

Jonathon N. Brownell, Esquire
Paterson, Gibson, Noble & Brownell
26 State Street
Montpelier, Vermont 05602

Peter S. Paine, Jr., Esquire
Clearly, Gottleib, Steen & Hamilton
52 Wall Street
New York, New York 10005

J. Eric Anderson, Esquire
Fitts and Olson
16 High Street
Brattleboro, Vermont 05301

William H. Ward, Esquire
Assistant Attorney General
Office of the Attorney General
State Capitol Building
Topeka, Kansas 66612

Donald W. Stever, Jr., Esquire
Office of the Attorney General
State House Annex
Concord, New Hampshire 03301

Chairman, Vermont Public Service
Corporation
Seven School Street
Montpelier, Vermont 05602

John W. Stevens, Director
Conservation Society of Southern
Vermont
Post Office Box 256
Townshend, Vermont 05353

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

cc w/enclosures and cy of VYNPC
ltr dtd 12/20/73:
Mr. Hans L. Hamester
ATTN: Joan Sause
Office of Radiation Programs
Environmental Protection Agency
Room 647A East Tower, Waterside Mall
401 M Street, S. W.
Washington, D. C. 20460

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

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

CHANGE NO. 15 TO TECHNICAL SPECIFICATIONS

INTRODUCTION

By a letter dated May 30, 1973, Vermont Yankee Nuclear Power Corporation (VYNPC) proposed changes to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would correct errors and inadequacies. As discussed in our Safety Evaluation for Change No. 13 dated January 17, 1974, to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station, the reissuance of the Technical Specifications will be made by sections. This Safety Evaluation will review "Limiting Conditions for Operation" and "Surveillance Requirements" for Sections 3.5 and 4.5, "Core and Containment Cooling Systems", Sections 3.6 and 4.6, "Reactor Coolant System", and Sections 3.7 and 4.7, "Station Containment Systems".

DISCUSSION

Sections 3.5 and 4.5

Minor word and organization changes were made for clarification in subsections A, B, C, D, and E. In Specification 3.5.F, an inoperable relief valve is acceptable for only 24 hours rather than 30 days and more than one inoperable relief valve is unacceptable. The resulting surveillance requirements have been modified accordingly in Specification 4.5.F. The change was necessary to reflect the new accident analysis performed by Vermont Yankee which indicates that all relief valves are required during a transient accident rather than only three of the four relief valves. Vermont Yankee is currently performing an analysis to determine the power reduction that would be necessary to allow a thirty-day operation with one relief valve inoperable and have the same safety margin as determined by the previous Safety Evaluation. The revised Safety Evaluation was required by the revision of the scram reactivity curve for the control rods. These changes are consistent with the discussion contained in our November 16, 1973 letter regarding scram reactivity insertion analysis.

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Minor word and organization changes were made for clarification in subsections G, H, I, J, and K. Bases were added for Specifications 3.5.D and 4.5.D and modified to reflect analysis change for Specification 3.5.F.

Sections 3.6 and 4.6

Minor word and organization changes were made for clarification in subsections A, C, D, and E. In Specification 3.6.B.1, the allowable steady state radioiodine concentration in the reactor coolant has been reduced from 20 uCi per ml of total iodine to 1.1 uCi per gram of I-131 dose equivalent. The results reflect a reanalysis by the Regulatory staff using site related meteorology, current calculational methods, and dose related iodine concentrations rather than gross iodine values. The bases have been changed to reflect the new analysis and parameters used. In Specification 4.6.B.1, the surveillance of the reactor coolant for radioiodine has been revised to reflect a means for surveillance of possible transient radioiodine spike conditions during startups and shutdowns. The bases have been changed to discuss the purpose of these added surveillance requirements. In Specifications 3.6.B.2 through 4, conductivity and chloride concentrations were changed to reflect Regulatory requirements stated in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors", for freshwater-cooled BWRs. The bases have been modified accordingly. In Specification 4.6.B.3.b, the sampling frequency has been increased to every four hours if the conductivity monitor is inoperable. The Regulatory position is that a conductivity monitor should be operable at all times during reactor operation as stated in Regulatory Guide 1.56. As an interim measure until Vermont Yankee can obtain and install a second conductivity monitor for redundancy, the foregoing specification for sampling has been incorporated. The Regulatory staff has requested from Vermont Yankee a schedule for purchasing and installing of the redundant conductivity monitor.

In Specification 3.6.F, the limiting conditions for operation of the jet pumps have been revised to reflect the Regulatory review of jet pump operation at other BWR plants. The specifications are consistent with those imposed on other jet pump BWR plants. In Specification 4.6.F, surveillance specifications have been added to complement the changes in Specification 3.6.F. The bases have been revised to reflect the reasons for these changes.

Specification 3.6.G has been added to indicate allowable operation with mismatch flow from the recirculation pumps. Specification 4.6.G provides the surveillance required in the determination of recirculation pump speeds. Bases have been added to give reasons for these specifications.

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Sections 3.7 and 4.7

Major modifications have been made to Specifications 3.7.A and 4.7.A. The surveillance required to assure primary containment integrity has been revised to remove all surveillance required by the recently adopted Appendix J to 10 CFR Part 50. Only those surveillance requirements not specified by Appendix J have been retained. Limiting conditions for operation have been added to reflect containment leakage rates assumed in the accident analysis. Minor changes to the bases have been made to reflect the nomenclature used by Appendix J. The limiting conditions for operation for the pressure suppression chamber-reactor building vacuum breakers have had minor changes made while an additional surveillance requirement has been added for each refueling outage. A major revision has been made for limiting conditions of operation and surveillance requirements for the pressure suppression chamber-drywell vacuum breakers. These changes reflect the Regulatory review of this system and the modifications to be made by June 1974 on the system surveillance by Vermont Yankee. Since the proposed surveillance system has not been made operational at this time, interim surveillance requirements are provided in the letter authorizing the Technical Specification change that is being concurrently issued with this Safety Evaluation for use by Vermont Yankee until the system becomes operational. The bases reflect the changes incorporated into these specifications. These changes reflect the discussion contained in our September 5, 1973 letter regarding a vacuum breaker surveillance system. The oxygen concentration limit in Specification 3.7.A.5 has been reduced from 5 percent to 4 percent consistent with the Regulatory review of the containment air dilution (CAD) system as discussed in our November 5, 1973 letter. Additional specifications will be required when the CAD system becomes operational after the next refueling outage in August 1974.

Major modifications to Specifications 3.7.B and 4.7.B have been made to reflect Regulatory requirements on systems used to reduce the consequences of postulated accidents. Since credit has been taken in the safety analysis for operation of the standby gas treatment system (SGTS), the limiting conditions for operation and surveillance requirements have been revised to meet the intent of Regulatory Guide 1.52 for an operating SGTS. The bases have been modified accordingly.

Minor word and organizational changes have been made for clarification in subsections C and D.

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CONCLUSIONS

On the basis of our evaluation, we have concluded that the changes proposed by VYNPC, as modified, and the changes necessary to meet Regulatory requirements, do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The Technical Specifications should be reissued as proposed by Vermont Yankee and modified by the Regulatory staff for Sections 3.5, 3.6, 3.7, 4.5, 4.6, and 4.7, including Bases. The remaining sections of the Technical Specifications shall be reissued as our review of proposed changes and Regulatory requirements is completed.

/s/

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

/s/

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

Date: JAN 28 1974

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3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling subsystems.

Objective:

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

Specification:

A. Core Spray and Low Pressure Coolant Injection

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core and containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Coolant Injection

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

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each refueling outage
b. Flow Rate Test (recirculate to torus) Core spray pumps shall deliver at least 3000 gpm against a system head of 120 psig. Three RHR (LPCI) pumps	Each refueling outage

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding fifteen days unless such subsystem is sooner made operable, provided that during such fifteen days all active components of the other core spray subsystem, the LPCI subsystems and the diesel generators required for operation of such components if no external source of power were available shall be operable.

3. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty

4.5 SURVEILLANCE REQUIREMENT

Item	Frequency
shall deliver 21600 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.	
c. Pump and Motor Operated Valve Operability	once/month

2. When it is determined that one core spray subsystem is inoperable the operable core spray subsystem and the LPCI subsystems and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3. When it is determined that one of the LPCI pumps is inoperable, the remaining active components of the LPCI and containment cooling subsystems, both core spray subsystems

3.5 LIMITING CONDITION FOR OPERATION

days unless such pump is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be operable.

4. From and after the date that a LPCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

B. <u>Containment Spray Cooling Capability</u>

1. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a

4.5 SURVEILLANCE REQUIREMENT

and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

4. When it is determined that a LPCI subsystem is inoperable, both core spray subsystems, the containment cooling subsystem, and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and daily thereafter.

B. <u>Containment Spray Cooling Capability</u>

Surveillance of the drywell spray loops shall be performed as follows:

During each five year period, an air test shall be performed on the drywell spray headers and nozzles.

3.5 LIMITING CONDITION FOR OPERATION

maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.

2. If this requirement cannot be met an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in Specifications 3.5.C.2, and 3.5.C.3 below, both RHR service water subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the RHR service water subsystem pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR service water subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT

C. Residual Heat Removal (RHR) Service Water System

Surveillance of the RHR service water system shall be performed as follows:

1. RHR service water subsystem testing:

   a. Pump and motor operated valve operability shall be tested every three months.

   b. Each RHR service water pump shall be tested after pump maintenance and every three months. Each pump shall deliver at least 2700 gpm and a pressure of at least 70 psia shall be maintained at the RHR heat exchanger service water outlet when the corresponding pairs of RHR service water pumps and station service water pumps are operating.

2. When it is determined that one RHR service water pump is inoperable, the remaining components of that subsystem and the other RHR service water subsystem shall be demonstrated to be operable immediately and daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that one RHR service water subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other RHR service water subsystem, both core spray subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

D. Station Service Water and Alternate Cooling Tower Systems

1. Except as specified in Specifications 3.5.D.2 and 3.5.D.3, both station service water subsystem loops and the alternate cooling

4.5 SURVEILLANCE REQUIREMENT

3. When one RHR service water subsystem becomes inoperable, the operable subsystem and the diesel generators required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.

D. Station Service Water and Alternate Cooling Tower Systems

Surveillance of the station service water and alternate cooling tower systems shall be performed as follows:

1. Pump and motor operated valve operability shall be tested every six months and whenever the plant is shutdown, but not

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
tower shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.	more than every three months. Flow rate test each station service water pump after pump maintenance and every three months. Each pump shall deliver at least 2700 gpm against a TDH of 250 feet.
2. From and after the date that one of the station service water subsystems is made or found inoperable for any reason, reactor operation is permissible only during the succeeding 15 days unless such subsystem is made operable, provided that during such 15 days all other active components of the station service water and alternate cooling tower system are operable.	2. When it is determined that one station service water subsystem is inoperable, the remaining operable components of that subsystem and the other station service water subsystem and the alternate cooling tower shall be demonstrated to be operable immediately and daily thereafter.
3. From and after the date that the alternate cooling tower subsystem or both station service water subsystems are made or found inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such subsystems are made operable, provided that during such 7 days all other active components of the other subsystem are operable.	3. When the alternate cooling tower or both station service water subsystems become inoperable, the operable subsystem and diesel generators required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.
4. If the requirements of Specification 3.5.D cannot be met an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	

3.5 LIMITING CONDITION FOR OPERATION

E. High Pressure Coolant Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig and prior to reactor startup from a cold condition:

   a. The HPCI system shall be operable.

   b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

2. From and after the date that the HPCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the automatic depressurization subsystems, the core spray subsystems, the LPCI subsystems, and the RCIC system are operable.

3. If the requirements of Specification 3.5.E cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI systems shall be performed as follows:

1. Testing:

Item	Frequency
Simulated Automatic Actuation Test	once/operating cycle
Pump operability	once/month
Motor operated valve operability	once/month
Flow rate test (recirculate to torus). The HPCI system shall deliver at least 4250 gpm to the reactor vessel for a range in reactor pressure of 1120 psig to 150 psig.	once/operating cycle

2. When it is determined that HPCI subsystem is inoperable, the LPCI subsystems, the core spray subsystems, the automatic depressurization subsystem, and the RCIC system shall be demonstrated to be operable immediately. The automatic depressurization valves and the RCIC system shall be demonstrated to be operable daily thereafter.

   NOTE: Automatic depressurization system operability shall be demonstrated by performing a functional test of the trip system logic.

3.5 LIMITING CONDITION FOR OPERATION

F. Automatic Depressurization System

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

2. From and after the date that one of the four relief valves of the automatic depressurization subsystem are made or found to be inoperable when the reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI subsystem is operable.

4.5 SURVEILLANCE REQUIREMENT

F. Automatic Depressurization System

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

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

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

3.5 LIMITING CONDITION FOR OPERATION

3. If the requirements of Specification 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 100 psig within 24 hours.

G. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in Specification 3.5.G.2 below, the RCIC system shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that the RCIC system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI system are operable.

4.5 SURVEILLANCE REQUIREMENT

G. Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC system shall be performed as follows:

1. Testing

Item	Frequency
Pump operability	Once/month
Motor operated valve operability	Once/month
Flow rate test. The RCIC shall deliver at least 400 gpm to the reactor vessel at normal operating pressure.	After major pump maintenance and every three months
Simulated automatic actuation test (testing valve operability)	Each refueling outage

2. When it is determined that the RCIC system is inoperable, the HPCI system shall be demonstrated to be operable immediately and daily thereafter.

3.5 LIMITING CONDITIONS FOR OPERATION

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the standby diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the low pressure core cooling and containment cooling subsystems connected to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENTS

H. Minimum Core and Containment Cooling System Availability

1. During reactor operation, when it is determined that one of the standby diesel generators is inoperable, all low pressure core cooling and containment cooling service water systems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

2. During a refueling outage, the surveillance requirements of Specification 4.5.H.1 shall be performed weekly.

3.5 LIMITING CONDITIONS FOR OPERATION

4. During a refueling outage, refueling operation may continue with one core spray system, the LPCI system or one of the emergency diesel generators inoperable for a period of thirty days.

I. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5 SURVEILLANCE REQUIREMENTS

I. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI subsystem or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

3.5 LIMITING CONDITIONS FOR OPERATION

J. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1.

K. Local LHGR

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

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

$LHGR_d$ = Design LHGR = 18.5 KW/ft

$\Delta P/P$ max = Maximum power spiking penalty = 0.038

LT = Total core length = 12 ft.

L = Axial position above bottom of the core

4.5 SURVEILLANCE REQUIREMENTS

J. Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

K. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

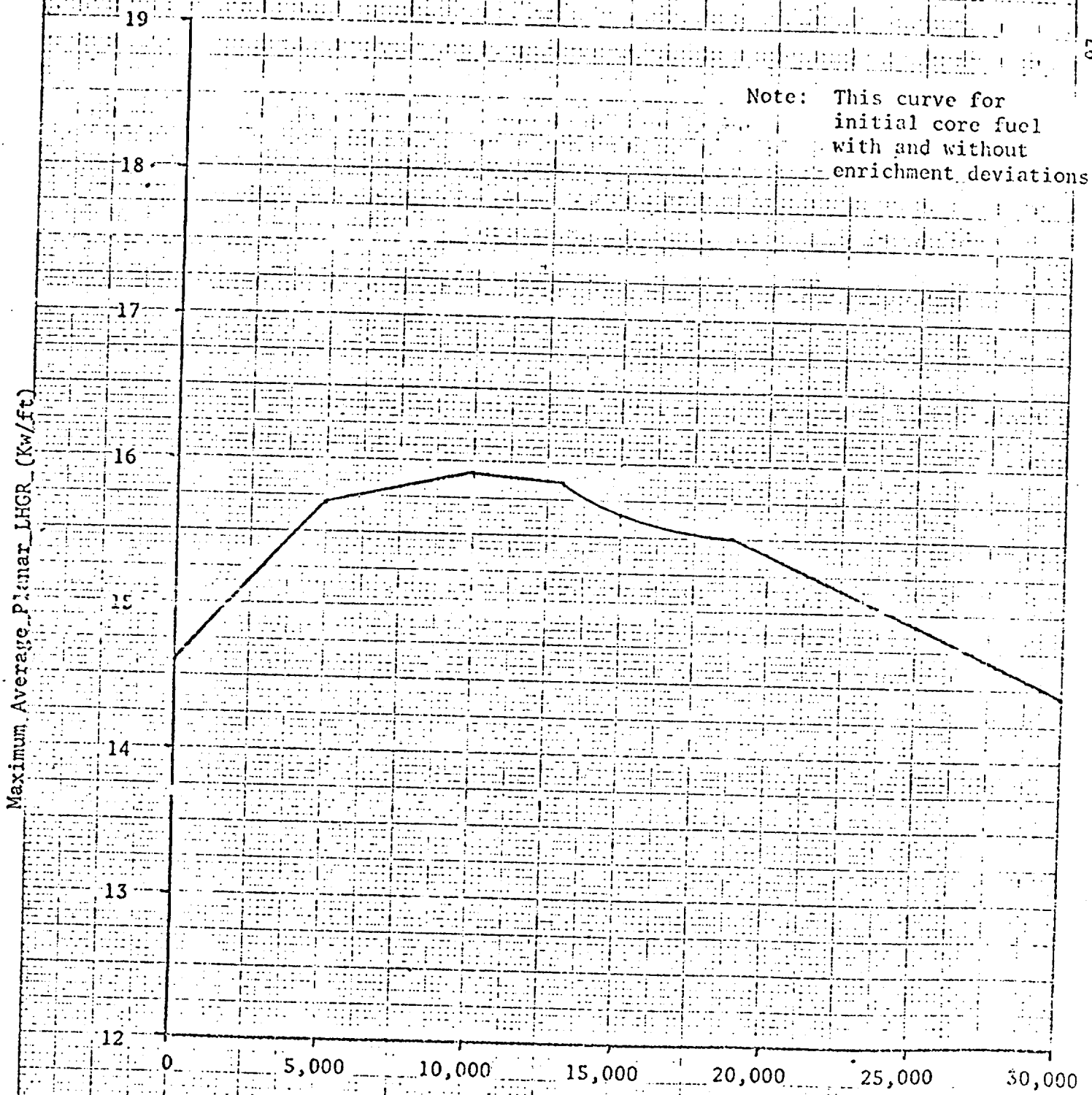

Figure 3.5.1.A Maximum Allowable Planar LHGR

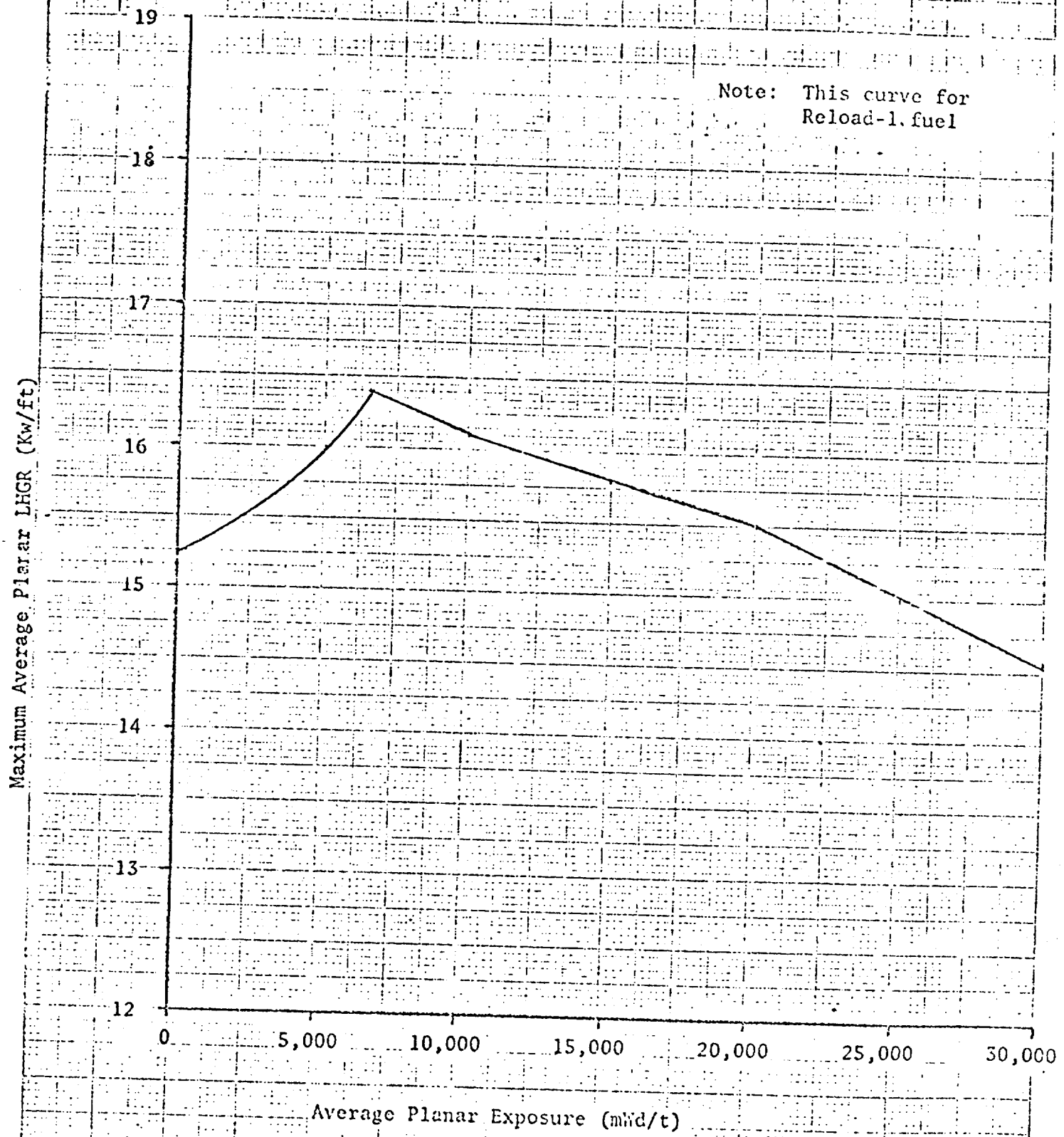

Figure 3.5.1.B Maximum Allowable Planar LHGR

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss of coolant analysis, any of the following cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to substantially below the melting point to assure that core geometry remains intact and to limit clad metal-water reaction: either of the two Core Spray subsystems or either of the two LPCI subsystems (a subsystem consists of two LPCI pumps and the LPCI injection valve powered from the same electrical source).

Limiting conditions specify three of the four standby core cooling subsystems operable, thus a margin of two subsystems over and above the one considered necessary in the accident analysis are available.

Each Core Spray subsystem has been shown, in a full scale test mockup of the VYNPS system, to exceed the minimum cooling requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor prior to reaching rated core spray conditions.

The LPCI system is also designed to provide sufficient core cooling in the event of a loss of coolant accident. This system is completely independent of the core spray cooling system. A single LPCI subsystem (2 LPCI pumps) provides adequate cooling for break areas of approximately 0.15 square feet up to and including 4.2 square feet, the latter being the double-ended recirculation line break, without any assistance from any other system.

3.5 (cont'd)

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference Section 14.6.3.3.2 FSAR.

The containment cooling subsystem consists of two sets of 2 RHR service water pumps, 1 heat exchanger and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger and 1 RHR pump has sufficient capacity to perform the cooling function. Whenever one containment cooling subsystem becomes inoperable, the remaining sybsystem will be tested daily.

D. Station Service Water and Alternate Cooling Tower Systems

The station service water subsystems and the alternate cooling tower system provide alternate heat sinks to dissipate residual heat after a shutdown or accident. Each station service water subsystem and the alternate cooling tower subsystem provides sufficient heat sink capacity to perform the required heat dissipation. The alternate cooling tower subsystem will provide the necessary heat sink in the event both station service water subsystems become incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond.

E. High Pressure Coolant Injection System

The high pressure coolant injection system (HPCIS) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray cooling subsystems can protect the core.

The HPCIS meets this requirement without the use of outside power. For the pipe breaks for which the HPCIS is intended to function the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference subsection 6.5.2.2 of the FSAR.

F. Automatic Depressurization System

The relief valves of the automatic depressurization system are a backup to the HPCIS. They enable the core spray cooling system or LPCI to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. Either of the two core spray cooling systems or LPCI provide sufficient flow of coolant to prevent clad melting. All four relief valves are included in the automatic pressure relief system. Of these four, only two are required to provide sufficient capacity for the automatic pressure relief system.

3.5 (cont'd)

G. Reactor Core Isolation Cooling System

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

H. Minimum Core and Containment Cooling System Availability

The core cooling and the containment cooling sybsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that adequate cooling capacity is available by precluding any combination of inoperable components from fulfulling the core and containment cooling function. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is in a cold shutdown condition and there is no potential for draining the reactor vessel.

I. Maintenance of Filled Discharge Pipe

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

J. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 even considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a 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 ± 20°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 temperatures are within the IAC limit.

The maximum average planar LHGR shown in Figures 3.5.1.A and 3.5.1.B is the same as that shown on the curve labeled "Ω" (Omega) on Figures 1-G and 2-G in the General Electric letter of J. A. Hinds to V. A. Moore, "Plant Evaluations with GEGAP-III," dated December 12, 1973, based on calculations employing the models described in the General Electric reports NEDM-10735 as modified by the General Electric report NEDO-20181 and the aforementioned General Electric letter of December 12, 1973.

3.5 (cont'd)

K. Local LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI

During normal plant operation, manual tests of operable pumps and valves shall be conducted monthly to demonstrate operability.

During each refueling shutdown, tests (as summarized below) shall be conducted to demonstrate proper automatic operation and system performance.

Periodic testing at the intervals specified above will demonstrate that all components which do not operate during normal conditions will operate properly if required.

The automatic actuation test will be performed by simulation of high drywell pressure or low-low water level. The starting of the pump and actuation of valves will be checked. The normal power supply will be used during the test. Testing of the sequencing of the pumps when the diesel generator is the source of power will be checked during the testing of the diesel. Following the automatic actuation test, the flow rate will be checked by recirculation to the suppression chamber. The pump and valve operability checks will be performed by manually starting the pump or activating the valve. For the pumps, the pump motors will be run long enough for them to reach operating temperatures.

B. and C. Containment Spray Cooling Capability and RHR Service Water Systems

The periodic testing intervals specified in Specifications 4.5.B. and C. will demonstrate that all components will operate properly if required. Since this is a manually actuated system, no automatic actuation test is required. The system will be activated manually and the flow checked by an indicator in the control room.

Once every five years air tests will be performed to assure that the containment spray header nozzles are operable.

D., E. and F. Station Service Water and Alternate Cooling Tower Systems and High Pressure Coolant Injection and Automatic Depressurization System

The testing intervals for the HPCI system will demonstrate that the system will operate if required. The automatic depressurization system is tested during refueling outages to avoid an undesirable blowdown of the reactor coolant system.

The HPCI automatic actuation test will be performed by simulation of the accident signal. This test will be followed by a flow rate test in which water is recirculated to the condensate storage tank.

4.5 (cont'd)

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

G. Reactor Core Isolation Cooling System

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

H. Minimum Core and Containment Cooling System Availability

Immediate testing followed by daily tests of all low pressure core cooling subsystems and containment cooling service water systems including the operable standby diesel generator upon determination of one inoperable diesel generator adequately demonstrates the availability of core and containment cooling subsystems.

I. Maintenance of Filled Discharge Pipe

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

J. and K. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal Limitations

1. The average rate of reactor coolant temperature change during normal heat-up or cooldown shall not exceed 100°F/hr when averaged over a one-hour period. However, a step reduction of 240°F is permissible if shell flange to shell temperature differential does not exceed 140°F.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6.1.

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 70°F.

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken in a 45-minute period is less than 5°F.

   a. reactor vessel shell
   b. reactor vessel shell flange
   c. recirculation loops A and B

2. Reactor vessel shell temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

3.6 LIMITING CONDITIONS FOR OPERATION

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

B. Coolant Chemistry

1. The steady state radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water.

4.6 SURVEILLANCE REQUIREMENTS

3. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program shall as a minimum conform to ASTME 185. The monitors and samples shall be removed and tested during the third refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 3.6.1.

B. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 uCi/sec, whichever is greater, during steady state reactor operation, a reactor coolant sample shall be taken and analyzed for radioactive iodines.

   b. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

3.6 LIMITING CONDITIONS FOR OPERATION

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in Specification 3.6.B.3:

Conductivity	5 umho/cm
Chloride ion	0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 umho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, in the reactor coolant water for the first 24 hours after placing the reactor in the power operating condition.

4. Except as specified in Specification 3.6.B.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hours.

Conductivity	5 umho/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.B. is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENTS

c. Whenever the steady state radioiodine concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.6.B.1, a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.

d. Whenever the steady state radioiodine concentration of prior operation is greater than 10 percent of Specification 3.6.B.1, a sample of reactor coolant shall be taken prior to any reactor startup and analyzed for radioactive iodines of I-131 through I-135 as well as the coolant sample and analyses required by Specification 4.6.B.1.c above.

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken every four hours and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

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

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

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable as required by Specification 3.5.F.

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

4.6 SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

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

D. Safety and Relief Valves

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

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

3.6 LIMITING CONDITION FOR OPERATION

E. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

F. Jet Pumps

1. Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

E. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

F. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3.6 LIMITING CONDITION FOR OPERATION

G. Recirculation Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If Specification 3.6.G.1 cannot be met, one recirculation pump shall be tripped.

4.6 SURVEILLANCE REQUIREMENT

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

G. Recirculation Pump Flow Mismatch

Recirculation pumps speed shall be checked daily for mismatch.

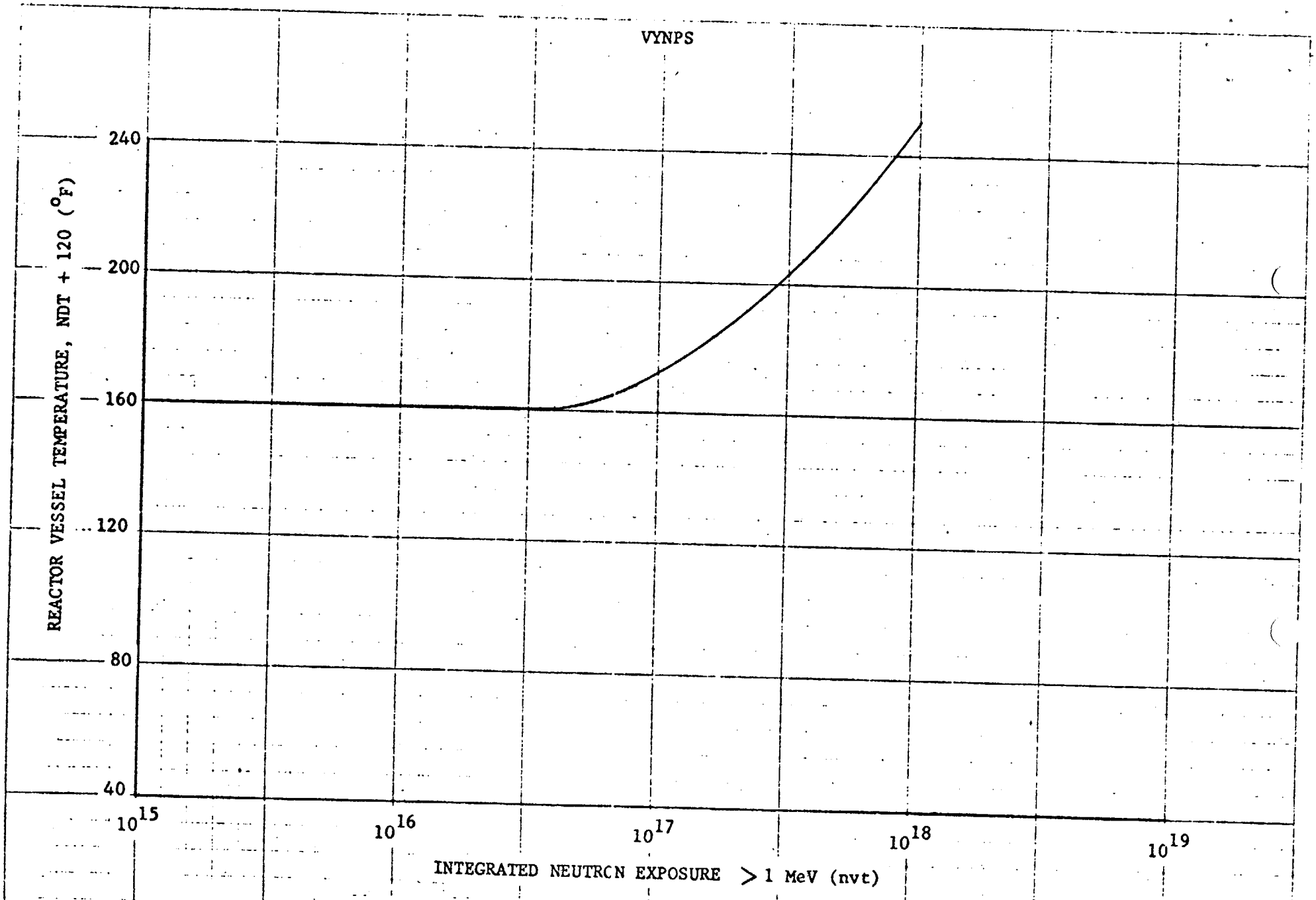

Figure 3.6.1 Minimum Reactor Pressurization Temperature

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 meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric	Closure Head: 3% of each meridional 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 & Surface	100% of the circumference of the safe-end weld; 25% of all safe-end welds
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
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

TABLE 4.6.1 (CONT'D)

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
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
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
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.

TABLE 4.6.1 (CONT'D)

CATEGORY	EXAMINATION AREA	METHOD OF EXAMINATION	(5-year period) EXTENT AND FREQUENCY OF EXAMINATION

SUPPLEMENTAL INSPECTION PROGRAM

The subgroup for the In-Service Inspection Code (ASME Section XI) is presently working on a program to include additional systems. These systems are those which perform safety functions to meet the criteria for the protection of public health and safety.

When the new additions to Section XI have been formally approved by the Main Committee of the American Society of Mechanical Engineers, Vermont Yankee's inspection program will comply with these additions.

3.6 & 4.6 REACTOR COOLANT SYSTEM

A. Thermal Limitations and Pressurization Temperature

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F per hour averaged over a period of one hour. This rate has been chosen, based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe station operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Detailed stress analyses were made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The specific conditions analyzed including 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F as well as other postulated occurrences for a 40 year life. Thermal stresses from this analysis combined with the primary load stresses fall within ASME Code Section III allowable stress intensities. The reactor vessel was built in accordance with Section III of the ASME Code.

During reactor operation, the temperature of the coolant in a idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the pump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel as discussed above.

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 mev) above 10^{17} nvt may increase the NDT temperature of the vessel base metal. Extensive tests have established the magnitude of changes in the NDT temperature as a function of the integrated neutron exposure. The FSAR presents pertinent test data for the type material (SA302B/SA533B) used as the base metal for this vessel.

The initial NDT temperature of the vessel shell material adjacent to the core is 40°F. The design life of the reactor vessel is 40 years and the maximum fast neutron fluence at the end of life is calculated to be 1.2×10^{17} nvt (>1 Mev).

3.6 & 4.6 (cont'd)

The "worst case" curve relating change in transition temperature to neutron fluence as presented in the FSAR was used to construct the "Minimum Reactor Pressurization Temperature" curve of Figure 3.6.1. This curve is based on an initial NDT of the vessel shell adjacent to the core. A 60°F margin based on the requirements of Section III of the ASME Boiler and Pressure Vessel Code, and a 60°F margin to account for the thickness effect of heavy section steel were added to 40°F to give the 160°F minimum temperature from initial operation to the time when the neutron fluence exceeds 5×10^{16} nvt. At that time the minimum temperature will increase steadily as the neutron fluence increases based on the "worst case" curve. After 40 years of operation the minimum operating temperature will be about 180°F.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The head flange and adjacent plate have an NDT of 10°F and are not subjected to any appreciable neutron fluence; therefore, the minimum temperature for bolting the vessel flange is 10°F + 60°F = 70°F.

Numerous data are available relating integrated flux and the change in nil-ductility transition temperature (NDTT) in various steels. The most conservative data has been used in Specification 3.6. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and, if necessary, correct the calculated data to determine an accurate NDTT.

In addition, vessel material samples will be located within the vessel to monitor the effect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal, heat affected zone metal, and standard specimens. These samples will receive neutron exposure more rapidly than the vessel wall material and, therefore, will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the shift in NDTT used in the specification is conservative.

B. Coolant Chemistry

A steady state radioiodine concentration limit of 1.1 uCi of I-131 dose equivalent per gram of water in the reactor coolant system can be reached if the gross radioactivity in the gaseous effluents are near the limit as set forth in Specification 3.8.A.2 or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the AEC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was

3.6 & 4.6 (cont'd)

calculated on the basis of the radioiodine concentration limit of 1.1 uCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3}$ sec/m^3 (Pasquill D and 0.33 m/sec equivalent) and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel. The attached graph, Figure 4.6.1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve.

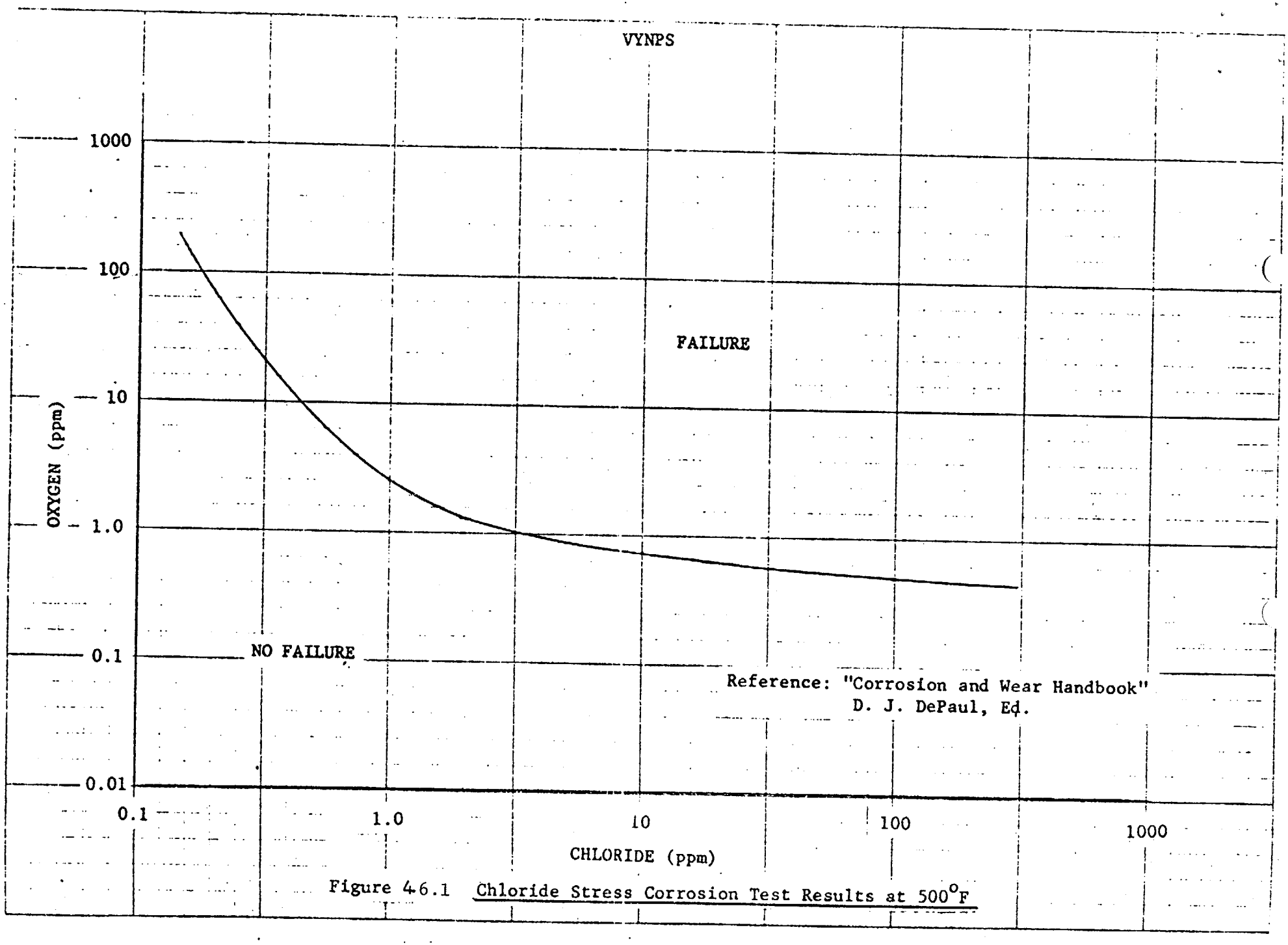

Figure 46.1 Chloride Stress Corrosion Test Results at 500°F

3.6 & 4.6 (cont'd)

When conductivity is in its proper normal range (approximately 10μmho/cm during reactor startup and 5μmho/cm during power operation), pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt, e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 5μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time when the conductivity exceed 5μmho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors", have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

3.6 & 4.6 (cont'd)

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

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

D. Safety and Relief Valves

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

E. Structural Integrity

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

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

3.6 & 4.6 (cont'd)

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

The in-service inspection program presented at this time is based on a thorough evaluation of present technology and state-of-the-art inspection techniques. The program will be continually re-evaluated as technology in the field of non-destructive inspection and equipment development. After five years, a new program will be presented to the AEC.

The interest of Vermont Yankee Nuclear Power Corporation in the development of new techniques for non-destructive testing of nuclear pressure boundaries is indicated by their participation in an Edison Electric Institute sponsored project. This project is primarily aimed at developing new techniques for continuous in-serivce inspection monitoring; namely, the accoustic emission and accoustic spectrometer techniques. The EEI program is also devoting some funding to the improved conventional ultrasonic inspection techniques.

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occured, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ±5%, the flow rates in both recirculation loops will be verified by main Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core f rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow throug the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measure. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

3.6 & 4.6 (cont'd)

The following factors form the basis for the surveillance requirements:

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

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

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

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

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

G. Recirculation Pump Flow Mismatch

The LPCI loop selection logic is described in the SAR, Section 6.4.4. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitations has been placed on the allowable variation in speed between the recirculation pumps.

3.6 & 4.6 (cont'd)

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   a. Maximum Water Temperature during normal operation - 90°F.
   b. Maximum Water Temperature during HPCI or RCIC test operation - 130°F and shall not be above 90°F for more than 24 hours.
   c. Minimum Water Volume - 68,000 cubic feet
   d. Maximum Water Volume - 78,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day. The interior painted surfaces above the water line of the pressure suppression chamber shall be inspected at each refueling outage.

2. The primary containment integrity shall be demonstrated as required by Appendix J to 10 CFR Part 50. The primary containment shall meet the containment acceptance requirements set forth in that appendix.

   a. Penetrations and seals listed in Table 4.7.1 shall be leak tested at 44 psig (Pa).

   b. Type C tests shall be performed on the isolation valves listed in Table 4.7.2a.

3.7 LIMITING CONDITIONS FOR OPERATION

3. Whenever primary containment is required, the total primary containment leakage rate shall not exceed 0.8 weight percent per day (L_a) at a pressure of 44 psig (P_a).

4. Whenever primary containment is required, the leakage from any one isolation valve shall not exceed 5 percent of the maximum allowable leak rate (L_a) at peak accident pressure (P_a) and the leakage from any one main steam line isolation valve shall not exceed 15.5 scf/hr at 44 psig (P_a).

5. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Two of two pressure suppression chamber-reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air-operated vacuum breakers shall be $\leq$ 0.5 psid. The self actuating vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breaker systems is made or found inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven (7) days unless such vacuum breaker system is sooner made operable, provided that the procedure does not violate containment integrity

4.7 SURVEILLANCE REQUIREMENTS

3. Prior to violating the integrity of a system outside the primary containment, which is connected to any valve listed in Table 4.7.2b, the isolation valves bounding the opening shall have Type C tests performed. If the opening cannot be isolated from the containment by two isolation valves which meet the acceptance criteria of Appendix J (10 CFR Part 50), a blank flange shall be installed on the opening.

4. The leakage from any one isolation valve shall not exceed 5% of Ltm. The leakage from any one main steam line isolation valve shall not exceed 11.5 scf/hr at 24 psig (Pt). Repair and retest shall be conducted to insure compliance.

5. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breaker systems and associated instrumentation including set point shall be checked for proper operation every three months.

   b. During each refueling outage, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specifications 3.7.A.5.a and each vacuum breaker shall be inspected and verified to meet design requirements.

3.7 LIMITING CONDITIONS FOR OPERATION

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specification 3.7.A.6.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

(1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

(2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.05 inch at all points along the seal surface of the disk.

(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.05 inch at all points along the seal surface of the disk.

b. Up to two (2) of the ten (10) suppression chamber - drywell vacuum breakers may be determined to be inoperable provided that they are secured, or known to be, in the closed position.

4.7 SURVEILLANCE REQUIREMENTS

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Periodic Operability Tests

Once each month and following any release of energy to the suppression chamber each suppression chamber - drywell vacuum breaker shall be exercised. Operability of valves, position switches, and position indicators and alarms shall be verified monthly and following any maintenance.

b. Refueling Outage Tests

(1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

(3) At least two (2) of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found such that Specification 3.7.A.6 could not be met, all vacuum breakers shall be inspected and deficiencies corrected.

3.7 LIMITING CONDITIONS FOR OPERATION

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

7. Oxygen Concentration

a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4 percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.7.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

8. If Specification 3.7.A cannot be met, an orderly shutdown shall be initiated immediately and the reactor shall be in a cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS

(4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

7. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITIONS FOR OPERATION

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, both circuits of the standby gas treatment system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA and charcoal filter banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

   b. The results of laboratory carbon sample analysis shall show $\geq$ 95% radioactive methyl iodide removal at a face velocity of 40 ft/min, 1 mg/m^3 inlet iodine concentration, 70% R.H. and 190°F.

   c. Fans shall be shown to operate at $\geq$ 90% design flow.

3. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit shall be operable.

4.7 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. At least once per operating cycle, the following conditions shall be demonstrated.

   a. Pressure drop across the combined HEPA and charcoal filter banks is less than 6 inches of water at 1500 cfm.

   b. Inlet heater input is at least 9 kW.

   c. Air distribution is uniform within 20% across HEPA filters and adsorbers.

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed initially and at least once per year for standby service or after every 720 hours of system operation and following painting, fire or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank.

   c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal filter bank.

   d. Each circuit shall be operated with the heaters on at least 10 hours every month.

   e. Test sealing of gaskets for housing doors downstream of the HEPA filters and adsorbers shall be performed at each test performed for compliance with Specification 4.7.B.2.a.

3.7 LIMITING CONDITIONS FOR OPERATION

4. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in Specifications 3.7.C.1(a) through (d), and compliance shall be completed within 24 hours thereafter.

C. Secondary Containment System

1. Integrity of the secondary containment system shall be maintained during all modes of plant operation except when all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.3.A is met and

4.7 SURVEILLANCE REQUIREMENTS

3. a. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

   b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

   c. When one circuit of the standby gas treatment system becomes inoperable the other circuit shall be demonstrated to be operable immediately and daily thereafter.

C. Secondary Containment System

1. Surveillance of secondary containment shall be performed as follows:

   a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind ($2 < u < 5$ mph) condition with a filter train flow rate of not more than 1500 cfm.

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
b. The reactor water temperature is below 212°F and the reactor coolant system is vented.	b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.
c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.	c. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than 1500 cfm, shall be demonstrated at least quarterly and at each refueling outage prior to refueling.
d. The fuel cask or irradiated fuel is not being moved in the reactor building.	
2. Core Spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.	2. The Core Spray and LPCI lower compartment openings shall be checked closed daily.
D. <u>Primary Containment Isolation Valves</u>	D. <u>Primary Containment Isolation Valves</u>
1. During reactor power operating conditions all isolation valves listed in Table 4.7.2 and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.	1. The primary containment isolation valves surveillance shall be performed as follows: a. At least once per operating cycle: (1) The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2.

3.7 LIMITING CONDITIONS FOR OPERATION

2. In the event any isolation valve specified in Table 4.7.2 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS

(2) The instrument line flow check valves shall be tested for proper operation.

b. At least once per quarter:

(1) All normally open power-operated isolation valves (except for the main steam line isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 50 percent of rated, trip main steam isolation valves (one at a time) and verify closure time.

At least twice per week the main steamline isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in Table 4.7.2 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

TABLE 4.7.1
PENETRATIONS AND SEALS SUBJECT TO TYPE B TESTING

Penetration Number	Identification	Number of Penetrations
X-7A, D	Main Steam Line A, D	4
X-9A, B	Feedwater Line A, B	2
X-11	HPCI Steam Line	1
X-12	Shutdown Cooling Supply	1
X-13A, B	RHR Return to Reactor	2
X-14	Supply to Reactor Water Cleanup	1
X-16A, B	Core Spray to Reactor	2
X-1	Equipment Access Hatch	1
X-3	Drywell Head Flange	1
X-4	Drywell Head Access Hatch	2
X-6	CRD Removal Hatch	1
SLH-A, H	Shear Lug Access Covers	8
X-202A, H & J, K	Vacuum Relief Access Covers	10
X-213A, B	Torus Drains	2
X-200A, B	Torus Manways	2

TABLE 4.7.2a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A,D & 2-86A,D)	4	4	5(note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9, 16-19-8)		2	10	Closed	SC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Closed	SC
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Closed	SC
3	Containment Purge Supply (16-19-23)		1	10	Closed	SC
3	Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22B)		3	NA	Closed	SC
5	Reactor Cleanup System (12-15, 12-18)	1	1	25	Open	GC
5	Reactor Cleanup System (12-68)		1	45	Open	GC
6	HPCI (23-15, 23-16)	1	1	55	Open	GC
6	RCIC (13-15, 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process

Table 4.7.2b

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES NOT SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A, B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A, B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A, B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A, B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A, B)		2	20	Open	GC
3	Containment Air Sampling System (109-75A, D; 1, 2 109- 76A, B)		10	5	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
4	RHR Reactor Head Cooling (10-32, 10-33)	1	1	25	Closed	SC
	Feedwater Check Valves (2-28 A, B)	2	2	NA	Open	Process
	Control Rod Hydraulic Return Check Valves (3-110, 3-113)	1	1	NA	Open	Process
	Reactor Head Cooling Check Valve (10-29)	1		NA	Closed	Process
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Process

Table 4.7.2 NOTES

1. Isolation signals are as follows:

Group 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Low-low reactor water level
2. High main steam line radiation
3. High main steam line flow
4. High main steam line tunnel temperature
5. Low main steam line pressure (run mode only)

Group 2: The valves in Group 2 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure

Group 3: The valves in Group 3 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor

Group 4: The valves in Group 4 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High reactor pressure

Group 5: The valves in Group 5 are closed upon low reactor water level.

Group 6: The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line. The signals indicating a steam line break for the respective steam line are as follows:

1. High steam line space temperature
2. High steam line flow
3. Low steam line pressure
4. High temperature in the main steam line tunnel (30 minute delay for the HPCI and the RCIC)

2. The closure time shall not be less than 3 seconds.

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during his period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable pressure suppression chamber pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 44 psig, which is below the design of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

3.7. A (cont'd)

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 90°F will assure that the 170°F limit is not approached.

Double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psig; the external design pressure is 2 psig.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows eight (8) operable valves, therefore, with two (2) valves secured, containment integrity is not impaired.

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of limit switches to provide fail safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.050" at all points along the seal surface of the disk. These switches are capable of transmitting the disk closed to open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundancy of components and circuits and, most importantly, the accessability of the valve lever arm and position reference external to the valve. The fail safe feature of the alarm circuits assures operator attention if a line fault occurs.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Code Allowable peak accident pressure is 62 psig.

The requirement to inert the containment is based on the recommendation of the Advisory Committee on Reactor Safeguards. This recommendation, in turn, is based on the assumption that several percent of the zirconium in the core will undergo a reaction with steam during the loss-of-coolant accident. This reaction would release sufficient hydrogen to result in a flammable concentration in the primary containment building. The oxygen concentration is therefore kept below 4% to minimize the possibility of hydrogen combustion.

General Electric has estimated that less than 0.1% of the zirconium would react with steam following a loss-of-coolant due to operation of emergency core cooling equipment. This quantity of zirconium would not liberate enough hydrogen to form a combustible mixture.

B. and C. Standby Gas Treatment System and Secondary Containment System

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 0.15 inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 100% capacity. If one standby gas treatment system circuit is inoperable, the other circuit must be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the plant is brought to a condition where the system is not required.

3.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

4.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment System

The water in the suppression chamber is used only for cooling in the event of an accident, i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 44 psig which would rapidly reduce to 27 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay.(1)

The design pressure of the drywell and absorption chamber is 56 psig.(2) The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

(1) Section 5.2 of the FSAR.
(2) 62 psig is the maximum allowable peak accident pressure for this design (56 psig) pressure.

4.7. A (cont'd)

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur over a 30-day period. Thus, these doeses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (La) to a value of 0.80%/day.

The maximum allowable test leak rate at the peak accident pressure of 44 psig (La) is 0.80 weight % per day. The maximum allowable test leak rate at the retest pressure of 24 psig (Lt) has been conservatively determined to be 0.59 weight percent per day. This value will be verified to be conservative by actual primary containment leak rate measurements at both 44 psig and 24 psig upon completion of the containment structure.

To allow a margin for possible leakage deterioation between test intervals, the maximum allowable operational leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 Lt. In addition, it is our intent to operate the primary containment structure at a slight positive pressure and to continuously monitor primary containment leakage. During normal plant operation only infrequent gas additions will be made to the primary containment and these will be made manually through a calibrated gas meter. Continuous primary containment temperature, pressure, and relative humidity data will be fed to the computer, which in turn will automatically calculate, by the absolute method, the actual weight of gas within the primary containment. This variation with time is the leakage rate and any change in this value is easily seen and permits corrective action to be taken to insure that the primary containment integrity is maintained. The reduced frequency for Type B tests is justified on the basis of this continuous leakage monitoring system.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetrations is conducted at the peak accident pressure of 44 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure.

4.7. A (cont'd)

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression chamber-reactor building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak - tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. As a result, a testing frequency of three months for operability is considered justified for this equipment. Inspections and calibrations are performed during the refueling outages, this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft^2. This is equivalent to one vacuum breaker open by three-eights of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a non-seated valve.

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber the rate of change of the suppression chamber pressure must not exceed a rate equivalent to the rate of leakage from the

4.7.A (cont'd)

drywell through a 1-inch orifice. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised monthly and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

Full Closed (Closed to $\leq$ 0.050" open)	2 White - On
Open (>0.050" open to full open)	2 White - Off

During each refueling outage, two drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in one-eighth of the design lifetime is extremely conservative.

Experience has shown that a weekly measurement of the oxygen concentration in the primary containment assures adequate surveillance of the primary containment atmosphere.

B. and C. Standby Gas Treatment System and Secondary Containment System

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 0.15 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leakage tightness of the reactor building, and performance of the standby gas treatment system. Functionally testing of initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

4.7 B & C (cont'd)

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after 2 years of operation in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging of the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval is reasonable. Such heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient. Air distribution tests will be conducted once during each operating cycle.

The in-place testing of charcoal filters is performed using Freon-112 or equivalent, which is injected into the system upstream of the charcoal filters. Measurements of the Freon concentration upstream and downstream of the charcoal filters is made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system.

High-efficiency particulate filters are installed before and after the charcoal filter to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP as testing medium.

The efficiencies of the particulate and charcoal filters are sufficient to prevent exceeding 10 CFR 100 limits for the accidents analyzed. The analysis of the loss-of-coolant accident assumed a charcoal filter efficiency of 90% and a particulate efficiency of 95%. Hence requiring efficiencies of 95% for the charcoal filters and 99% for the particulate filters provides adequate margin.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, testing for retention capacity in terms of microcuries of iodine per gram of charcoal will be demonstrated. This will be done by removing one adsorber tray from the system and using the adsorbent from one bed after mixing to obtain at least two samples equivalent to the bed in thickness and diameter. Also test for gasket leakage will be performed to meet ANSI N101.1. These tests will normally be performed at least every year.

4.7. D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. There, this isolation valve closure time is sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of 5 seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10 CFR 100 limits are not exceeded. Redundant valves in each line insure that isolation will be effected applying the single failure criteria.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. A program for periodic testing and examiniation of the flow check valves in these lines is performed similar to that described in Amendment No. 23, Millstone Unit 1, Docket 50-245.

Docket File

JAN 17 1974

Docket No. 50-271

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Albert A. Cree, President
77 Grove Street
Rutland, Vermont 05701

Change No. 13
License No. DPR-28

Gentlemen:

Your letter dated May 30, 1973, proposed changes to the Technical Specifications of Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would correct errors and inadequacies. Concurrent with your request, meetings were being held between our respective staffs regarding the installation of the augmented off-gas system (AOG), which was connected during the recent outage, and associated technical specifications. Additional modifications to the drywell vacuum breaker system and the containment air dilution system require technical specification changes. Meetings were held at the Vermont Yankee plant site in October between our staffs to discuss a reissuance of the Technical Specifications and the changes necessary to meet Regulatory requirements. A meeting was held at our Bethesda office between our staffs to discuss changes to the Technical Specifications necessary to incorporate as low as practicable effluent limits and to operate the AOG system following the check out period as stated in our August 29, 1973 letter.

Due to the extensive changes being made to the Technical Specifications, the reissuance of the Technical Specifications appended to Facility License No. DPR-28 will be made by sections consistent with our review and in numerical order with the current specifications. Several areas of the enclosed technical specifications will require further revision as soon as the systems can be modified at the next refueling outage. These areas are discussed in our related Safety Evaluation which is enclosed. On the basis of our review, we have concluded that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed by replacing Sections 1.0, 1.1, 1.2, 2.1, 2.2, 3.1, 3.2, 3.3, 3.4, 4.1, 4.2, 4.3, and 4.4 (pages 1 through 83) in their entirety with the enclosed sections.

As discussed between our respective staffs and stated in our related Safety Evaluation, Vermont Yankee is investigating modifications to the following systems which are to be made to the Vermont Yankee plant prior to or during the next refueling outage. These modifications will result in a need for additional changes to these approved Technical Specification changes. The affected systems include: (1) APRM System (Specification 3.1), (2) Relief and Safety Valve Settings (Specification 2.2), (3) Primary Containment Isolation for Low Turbine Condenser Vacuum (Specification 3.2), and (4) Off-Gas System Radiation Trip Settings (Specification 3.2). Another system being investigated by Vermont Yankee is the Control Room Ventilation System which will require changes in Specifications 3.2 and 4.2 at a later time.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Technical Specifications
(pages 1 through 84)

cc w/enclosures: See next page

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cc: Mr. Lawrence E. Minnick, Vice President
Vermont Yankee Nuclear Power Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

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

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

Richard E. Ayres, Esquire
David Schoenbrod, Esquire
National Resources Defense Council, Inc.
15 West 44th Street
New York, New York 10036

Honorable Kimberly B. Cheney
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

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

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

Jonathon N. Brownell, Esquire
Paterson, Gibson, Noble & Brownell
26 State Street
Montpelier, Vermont 05602

Peter S. Paine, Jr., Esquire
Cleary, Gottleib, Steen & Hamilton
52 Wall Street
New York, New York 10005

J. Eric Anderson, Esquire
Fitts and Olson
16 High Street
Brattleboro, Vermont 05301

William H. Ward, Esquire
Assistant Attorney General
Office of the Attorney General
State Capitol Building
Topeka, Kansas 66612

Donald W. Stever, Jr., Esquire
Office of the Attorney General
State House Annex
Concord, New Hampshire 03301

Chairman, Vermont Public Servic
Corporation
Seven School Street
Montpelier, Vermont 05602

John W. Stevens, Director
Conservation Society of Souther
Vermont
Post Office Box 256
Townshend, Vermont 05353

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

cc w/enclosures and cy of
VY ltr dtd 5/30/73:
Mr. Hans L. Hamester
ATTN: Joan Sause
Office of Radiation Programs
Environmental Protection Agency
Room 647A East Tower, Waterside Mall
401 M Street, S. W.
Washington, D. C. 20460

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

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed by replacing Sections 1.0, 1.1, 1.2, 2.1, 2.2, 3.1, 3.2, 3.3, 3.4, 4.1, 4.2, 4.3, and 4.4 (pages 1 through 83) in their entirety with the enclosed sections.

As discussed between our respective staffs and stated in our related Safety Evaluation, Vermont Yankee is investigating modifications to the following systems which are to be made to the Vermont Yankee plant prior to or during the next refueling outage. These modifications will result in a need for additional changes to these approved Technical Specification changes. The affected systems include: (1) APRM System (specification 3.1), (2) Relief and Safety Valve Settings (specification 2.2), (3) Primary Containment Isolation for Low Turbine Condenser Vacuum (specification 3.2), and (4) Off-Gas System Radiation Trip Settings (specification 3.2). Another system being investigated by Vermont Yankee is the Control Room Ventilation System which will require changes in specifications 3.2 and 4.2 at a later time.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

CHANGE NO. 13 TO TECHNICAL SPECIFICATIONS

Introduction

By a letter dated May 30, 1973, Vermont Yankee Nuclear Power Corporation (VYNPC) proposed changes to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would correct errors and inadequacies. Concurrent with this request, meetings were being held between the VYNPC staff and the Regulatory staff regarding installation of the augmented off-gas system, modifications to the drywell vacuum breaker system, and additions to the containment air dilution system. Following these discussions, a reissuance of the Technical Specifications was determined to be appropriate in order to incorporate the proposed changes, system modifications, and other Regulatory requirements. The reissuance of the Technical Specifications will be made by sections consistent with our review and in numerical order with the current specifications. Several areas of the technical specifications will require further revision after the systems have been modified during the next refueling. Each of these areas will be discussed in the safety evaluation and modification of the system as discussed will be a requirement. This safety evaluation will review Section 1.0 "Definitions", Sections 1.1 and 1.2, "Safety Limit", and Sections 2.1 and 2.2, "Limiting Safety System Setting". The safety evaluation will also review "Limiting Conditions for Operation" and "Surveillance Requirements" for Sections 3.1 and 4.1, "Reactor Protection System", Sections 3.2 and 4.2, "Protective Instrumentation System", Sections 3.3 and 4.3, "Control Rod System", and Sections 3.4 and 4.4, "Reactor Standby Liquid Control System".

Discussion

Section 1.0

Changes to this section included a redefinition for abnormal occurrence which complies with the definitions given in Regulatory Guide 1.16,

"Reporting of Operating Information". An additional shutdown condition was added to clarify the definition. Other minor changes were incorporated for clarification.

Sections 1.1 and 1.2

A requirement for continuous monitoring of the water level was added to Specification 1.1.D. Minor word changes were made for clarification.

Sections 2.1 and 2.2

Peaking factor was changed to total peaking factor to provide a more accurate terminology. The IRM flux scram setting value has been added to Specification 2.1.A.2. Modifications shall be made to the APRM system so that the rated neutron flux will be indicated and a scram set point of less than or equal to 15 percent in the Startup and Refuel Modes shall be required. This change will be reflected in Specification 3.1, Table 3.1.1 as a revision after the system modification is completed. A terminology change was made in Specification 2.1.F. Changes to Specification 2.2.B will be required as discussed in our November 16, 1973 letter regarding scram reactivity insertion analysis for the current operating cycle. These changes shall result in lowering the relief and safety valve settings and minor changes to set point error allowances. Minor wording changes have been made in the Bases to reflect current analysis and specification changes.

Sections 3.1 and 4.1

In Specification 4.1.B, reference to Table 3.2.3 was changed to Table 3.2.5 for correction. In Table 3.1.1, APRM scram function in Refuel and Startup Mode was removed as a correction. Note 11 was added to Table 3.1.1 for the IRM scram function. The turbine condenser low vacuum scram has been removed from Table 3.1.1. Based upon the request by VYNPC, the Regulatory staff reviewed the need and purpose of this scram. We concluded that the scram was unnecessary and would not provide any protective function to the reactor system. The scram function could result in unnecessary delay during reactor startup as discussed by VYNPC in the proposed change dated May 30, 1973. The review of the turbine condenser system by the Regulatory staff did indicate a need for a primary containment isolation upon low turbine condenser vacuum in order to prevent possible loss of radioactive gases from the primary coolant through the condenser. Modifications to the instrumentation are necessary before such an isolation can be performed automatically upon loss of condenser vacuum. Such modifications are being investigated by VYNPC for incorporation during the next refueling outage. At that

time, a revision to Table 3.2.2 in Specification 3.2.B will be necessary to include this isolation function. Note 1.(c) to Table 3.1.1 has been changed by adding high flux SRM scram. The Bases for this specification has been modified to reflect the change. A new Note 7 has been added to indicate that the main steamline high radiation channel is shared by the Reactor Protection System and the Primary Containment Isolation System instrumentation. Note 8 indicates an alarm level of 1.5 times normal background for the main steamline high radiation channel. The trip setting for this channel has been reduced from 7 times to 3 times normal background at rated power in order to provide earlier scram of the reactor in case of gross fuel failure in the core. Table 4.1.1 has been expanded to Tables 4.1.1 and 4.1.2. These tables now reflect the frequency for functional tests (Table 4.1.1) and frequency for calibration (Table 4.1.2). Other changes to the tables have been made for clarification.

Sections 3.2 and 4.2

These sections have been reorganized for clarification. Specifications 3.2.F and 4.2.F, "Mechanical Vacuum Pump Isolation", have been added to these sections and removed from Specification 3.8 since this requirement is associated with a protective function and not effluent release function. Minor changes have been made to the wording of this specification to add a reactor shutdown requirement if isolation cannot be accomplished. Specification 3.2.B, "Surveillance Instrumentation", has been changed to Specification 3.2.G, "Post-Accident Instrumentation". Another section will be required in these specifications for the control room ventilation system. VYNPC has been informed of this need and is currently investigating the system for possible modifications and technical specification requirements. All trip level settings in the tables for these specifications have been changed to reflect absolute settings rather than an allowable range for clarification of intent. Changes to these settings, consistent with VYNPC requests and Regulatory staff review of the affected systems, have been made. Table 3.2.2 will be modified at a later date to add the trip function associated with a low turbine condenser vacuum as previously discussed above for Sections 3.1 and 4.1. For systems with a time delay relay, the actual relay to be tested has been added to the table by number for definite identification. Notes 7, 8, and 9 have been added to Table 3.2.2. Note 7 reflects the signal that closes the mechanical vacuum pump suction line isolation valve. Note 8 indicates channel sharing by the Reactor Protection and Primary Containment Isolation Systems. Note 9 indicates alarm setting on the main steamline high radiation channel. In Table 3.2.3, the Reactor Building Vent monitor trip setting has been

reduced from an arbitrary value to a value consistent with the effluent release limits given in Specification 3.8. The Bases has been modified to reflect this change. Table 3.2.4 has been modified to reflect new trip functions for the Off-Gas System Radiation monitors with the installation of the augmented off-gas system (AOG) during the recent outage. Note 3 to this table indicates the operator action required while the Bases has been modified to reflect the change. Automatic isolation for these trip settings is not currently available. VYNPC is investigating the equipment modifications necessary to make such isolation automatic and shall make such modifications during the next refueling. Additional specification changes will be necessary at that time. These trip settings reflect allowable effluent release rates at the air ejector for limiting accident doses in case of a system failure rather than for limiting normal effluent releases as previously required prior to AOG installation. Table 3.2.6 has been modified to indicate the range for the post-accident instruments and has had the torus air temperature monitor added. The instrument checks deleted by Change No. 2 dated July 28, 1972, have been reinstated and changed from once each shift to daily. In Tables 4.2.3 and 4.2.4, the functional test of the radiation monitors shall be performed monthly without qualification. In Table 4.2.5, the high water level in scram discharge volume was added. In Table 4.2.6, the torus air temperature was added. Note 8 was added for all Table 4.2 to restrict the need for functional tests and calibration to periods when the systems are required to be operable. Other minor word changes were made in the tables and bases for clarification.

Sections 3.3 and 4.3

Specification 3.3.B.3 has been modified to define the number of control rods that must be withdrawn at the time of the RSM failure to allow continuance of the reactor startup with another licensed operator. Specification 4.3.B.4 has been added as surveillance for Specification 3.3.B.4 and Specification 3.3.B.4.a) has been changed from 1.25% to 1.3% to reflect the new analysis given in the Bases. The need for the nuclear engineer in Specification 3.3.B.6 has been deleted. The surveillance requirements in Specification 4.3.C have been modified in the same manner as given in Change No. 10 dated October 19, 1973, for Quad-Cities Units 1 and 2, Docket Nos. 50-254 and 50-265. The changes reflect the request by VYNPC in their letter dated May 30, 1973. The special report required by the deleted surveillance requirement is being prepared by VYNPC and will be submitted to the AEC for use by the Regulatory staff. The modified specification requires that the

results of all future measurements on the control rod drives be submitted by VYNPC in the semiannual operating report to the AEC. Specifications 3.3.C.3 and 4 have been added to clarify the intent of Specification 3.3.C.

Sections 3.4 and 4.4

Minor editorial changes were made in these specifications.

Conclusions

On the basis of our evaluation, we have concluded that the changes proposed by VYNPC, as modified, and the changes necessary to meet regulatory requirements do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The Technical Specifications should be reissued as proposed by Vermont Yankee and modified by the AEC staff for Sections 1.0, 1.1, 1.2, 2.1, 2.2, 3.1, 3.2, 3.3, 3.4, 4.1, 4.2, 4.3, and 4.4, including Bases. All sections of the Technical Specifications shall be reissued as our review of proposed changes and regulatory requirements is completed.

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

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

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