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TO: Mr. Victor Stello

FROM: NSP
Minneapolis, Minnesota 55401
L. O. Mayer

DATE OF DOCUMENT: 08/30/77

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ENCLOSURE

License No. DPR-22 Appl for Amend: Tech Specs proposed change concerning requiring inservice inspection and testing to be performed in accordance with the examination and testing requirements set forth in Section XI of the ASME Code and Addenda...Notorized 08/30/77

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

August 30, 1977

Regulatory File Cy

RECEIVED SEP 1 1977 US NUCLEAR REGULATORY COMMISSION Mail Section

Mr Victor Stello, Director
Division of Operating Reactors
c/o Distribution Services Branch, DDC, ADM
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Stello:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request Dated August 30, 1977

Attached are three signed originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A, of Operating License DPR-22.

This change is submitted in response to a letter dated April 28, 1976 from Mr D L Ziemann, Chief, Operating Reactors Branch #2, USNRC. This letter requested Northern States Power Company to submit a License Amendment Request to modify the Technical Specifications to require inservice inspection and testing to be performed in accordance with the examination and testing requirements set forth in Section XI of the ASME Code and Addenda. The requested changes must be issued with an effective date of February 28, 1978.

This License Amendment Request has been reviewed by the Monticello Operations Committee and the Monticello Safety Audit Committee.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/DMM/deh

cc: J G Keppler
G Charnoff
MPCA--Attn: J W Ferman

Attachments

7724 50036

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50- 263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR- 22

(License Amendment Request Dated August 30, 1977)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production &
System Operation

On this 30th day of August, 1977, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST DATED AUGUST 30, 1977

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF OPERATING LICENSE
DPR-22

Pursuant to 10 CFR 50, Section 50.59, the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications.

The changes should be issued with an effective date of February 28, 1978.

1. Inservice Inspection and Testing Program

Proposed Change

Revise the Technical Specifications to require inservice inspection and testing to conform to the requirements of 10 CFR 50, Section 50.55a. Refer to the revised Limiting Conditions for Operation and Surveillance Requirements contained in Exhibit B.

These changes implement 10 CFR 50, Section 50.55a by:

a. Eliminating existing surveillance requirements which are redundant to the inspection and testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

b. Adding new section 3.13/4.13 which requires inspection and testing to conform to the requirements of 10 CFR 50, Section 50.55a. Beginning with the inspection period starting February 28, 1978 inservice inspection and testing will conform to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda where practical. The inservice inspection and testing program will be upgraded at 20 and 40-month intervals as required by 10 CFR 50, Section 50.55a. Requests for deviation from the requirements of Section XI of the ASME Code will be submitted to the Commission.

Reason for Change

This change is submitted in response to a letter dated April 28, 1976 from Mr. D. L. Ziemann, USNRC, to Mr. L. O. Mayer, NSP. This letter directed NSP to submit a License Amendment Request to modify the Technical Specifications to require inservice inspection and testing to be performed in accordance with the examination and testing requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

The proposed changes remove the conflicts between the current surveillance requirements and the requirements of Section XI of the ASME Code (1974 edition and addenda through Summer 1975). 10 CFR 50, Section 50.55a(g) requires, where practical, that the program of inservice inspection and testing conform to editions of the Code and addenda in effect no more than six months prior to the start of each inspection period. The 1974 edition of the Code and addenda through Summer 1975 are the latest version of Section XI to be placed into effect by the Commission. Generalized wording has been used in the revisions so that future changes to the program conforming to 10 CFR 50, Section 50.55a, can be made without the need for additional changes to the Technical Specifications.

The proposed wording in this change request generally conforms to the suggested wording contained in Mr. Ziemann's letter of April 28, 1976 with the following exceptions:

a. NRC Quality Groups have been used to classify components. Classifying components by ASME Code Class for purposes of inspection and testing could be done, however it may imply that these components satisfy other requirements of the ASME Code Classes. The majority of Monticello components were installed before Code Classes were adopted by the ASME Code.

b. The wording in the proposed changes states that inspection and testing will conform to Section XI of the ASME Code "... except where relief has been requested from the Commission pursuant to ..." rather than "... except where specific written relief has been granted by the NRC pursuant to ...." This wording change permits Section XI requirements which are impossible to comply with to be excluded from the testing program immediately upon discovery without violating the Technical Specifications. Following discovery of an impossible or impractical requirement, the Commission will be notified and information will be submitted to support our determination. If, as a result of thier review, the Commission rejects the proposed deviation they can order the necessary plant modifications or procedural changes. In the interim, the requested deviation will be assumed to be acceptable.

Reason for Change (Continued)

Deviations from the proposed wording contained in Mr. Ziemann's letter are necessary to permit a workable Section XI inspection and testing program to be backfit to the Monticello plant.

Safety Evaluation

This change is being submitted at the request of the Commission. The proposed changes will require future inservice inspection and testing to conform to the current ASME Section XI requirements where practical. This will permit improvements in inspection and testing techniques to be incorporated at regular intervals in the Monticello inservice inspection and testing program.

Implementation of these changes will substantially increase the scope of inservice inspection and testing at Monticello over what was originally contemplated. This will generally result in an enhancement of the inservice inspection and testing program. Therefore these changes do not involve a significant hazards consideration and do not adversely affect the common defense and security or the health and safety of the public.

2. Renumbering of Pages

Proposed Change

The proposed changes in (1) above delete Table 4.6.1 which is currently in the Technical Specifications (pages 124 - 129). We ask that existing pages 121A - 121C, 122, 122A - 122C, and 123 be renumbered as pages 122 - 129. Refer to Exhibit B which contains the proposed renumbered pages.

Reason for Change

This change would eliminate the gap in page numbering created by the deletion of Table 4.6.1.

Safety Evaluation

No changes to the Technical Specifications are involved. Only page numbers would be revised.

3. Table of Contents, List of Figures, and List of Tables

Proposed Change

Revise the Table of Contents, List of Figures, and List of Tables for the Appendix A Technical Specifications to reflect changes (1) and (2) above and to correct a number of minor errors. Refer to Exhibit B pages ii, iii, v,vii, and ix.

Reason for Change

These changes are required to update the Table of Contnents, List of Figures, and List of Tables.

Safety Evaluation

No changes to the Technical Specifications are involved.

EXHIBIT B

LICENSE AMENDMENT REQUEST DATED AUGUST 30, 1977

Exhibit B, attached, consists of revised and newly prepared pages of the Appendix A Technical Specifications as listed below. These pages incorporate the proposed changes.

Pages
ii
iii
v
vii
ix
88
89
90
94
96
97
98
99
100
101
102
103
104
105
106
107
108A
114
119
120
122 - 129 (renumbered pages 121A - 123)
135
136
137
138
146
151
152
189R (new page)
189S (new page)
189T (new page)

<u>LIST OF FIGURES</u>

LIST OF TABLES

3.0 LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

SPECIFICATION:

A. Normal Operation

1. The standby liquid control system shall be operable at all times when fuel is in the reactor and the reactor is not shutdown by control rods, except as specified in 3.4.B.

2. Each standby liquid control system pump shall be capable of delivering 24 gpm against a reactor pressure of 1275 psig.

3. The system pressure relief valves shall be operable with a setpoint between 1350 and 1450 psig.

4.0 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

SPECIFICATION:

A. The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once each operating cycle manually initiate one of the two standby liquid control systems and pump demineralized water into the reactor vessel. Both systems shall be tested and inspected in the course of two operating cycles.

2. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, provided that the component is returned to an operable condition within seven days.

B. Surveillance with Inoperable Components

When a component becomes inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter.

C. Volume—Concentration Requirements

The liquid poison tank shall contain a boron bearing solution that satisfies the volume—concentration requirements of Figure 3.4.1 and at all times when the standby liquid poison system is required to be operable the temperature shall not be less than the solution temperature presented in Figure 3.4.2. In addition, the heat tracing on the pump suction lines shall be operable whenever the room temperature is less than the solution temperature presented in Figure 3.4.2.

C. The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. <u>At least once per month</u> –

Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

Bases 3.4 and 4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of 900 ppm of boron in the reactor core in less than 125 minutes. 900 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3% Δk subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin for possible imperfect mixing of the chemical solution in the reactor water and dilution from the water in the cooldown circuit. A minimum net quantity of 1400 gallons of solution having a 21.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. The maximum net storage volume of the boron solution is 2895 gallons. (256 gallons are contained below the pump suction and, therefore, have not been used in the net quantities above.)

Boron concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability demonstrates that testing at a three-month interval is adequate to detect if failures have occurred.

Standby liquid control system components are inspected and tested in accordance with the requirements of 10 CFR 50, Section 50.55a(g). These requirements are delineated in Specification 4.13. This inspection and testing program, combined with the additional surveillance requirements contained in this section, provide a high degree of assurance that the standby liquid control system will perform as required when needed.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

3.0 LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling systems.

Objective:

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

Specification:

Low Pressure Core Cooling Capability

A. Core Spray System

1. Except as specified in 3.5.A.2., 3.5.A.3., and 3.5.A.5. below, both core spray subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212°F.

4.0 SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling systems.

Objective:

To verify the operability of the emergency cooling systems.

Specification:

Low Pressure Core Cooling Capability

A. Surveillance of the core spray system shall be performed as follows:

1. Routine Testing

   a. A simulated automatic actuation test shall be conducted each refueling outage.

   b. Core spray header Δp instrumentation shall be checked once each day, tested once each month, and calibrated once each 3-month period.

c. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

2. From and after the date that one of the core spray systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding fifteen days unless such system is sooner made operable, provided that during such fifteen days all active components of the other core spray system and the LPCI mode of the RHR system 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 both core spray systems are made or found to be inoperable for any reason, reactor

2. When it is determined that one core spray system is inoperable, the operable core spray system and the LPCI mode of the RHR system 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 system shall be demonstrated to be operable daily thereafter.

3. When it is determined that both core spray systems are inoperable, the LPCI mode of the RHR system and the

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
operation is permissible only during the succeeding seven days unless at least one of such systems is sooner made operable, provided that during such seven days all active components of the LPCI mode of RHR system and the diesel generators required for operation of such components (if no external source of power were available) shall be operable. 4. Each core spray system shall be capable of delivering 3,020 gpm against a reactor pressure of 130 psig. If this rate of delivery requirement cannot be met, the system shall be considered inoperable. 5. If the requirements of 3.5.A.1 - 3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212 °F within 24 hours.	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.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Low Pressure Coolant Injection (LPCI) Subsystem (LPCI mode of RHR system)

1. Except as specified in 3.5.B.2 and 3.5.B.3 below, the LPCI shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the LPCI pumps or admission valves is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump or admission valve 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 systems and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

4.0 SURVEILLANCE REQUIREMENTS

B. Surveillance of the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI mode of RHR system) shall be performed as follows:

1. Routine Testing

   a. A simulated automatic actuation test shall be conducted each refueling outage.

   b. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

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

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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. From and after the date that two of the LPCI pumps or admission valves are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such pumps or admission valves are made operable sooner, provided that during such seven days all active components of both core spray systems, 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 demonstrated to be operable at least once each day.	3. When it is determined that the LPCI subsystem is inoperable, both core spray systems, 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.
4. A maximum of one drywell spray loop (containment cooling mode of RHR) may be inoperable for 30 days when the reactor water temperature is greater than 212°F. If the loop is not returned to service within 30 days, the orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F.	
5. Each LPCI subsystem (RHR) pump shall be capable of delivering 4,000 gpm against a reactor pressure of 20 psig. If this	

rate of delivery requirement cannot be met, the pump shall be considered inoperable.

6. If the requirements of 3.5.B.1-4 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

Containment Cooling Capability

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in 3.5.C.2 and 3.5.C.3 below, both RHR service water system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the RHR service water system pumps is made or found to be inoperable for any reason,

Containment Cooling Capability

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

1. **Inservice inspection and testing** of components shall be conducted in accordance with Specification 4.13.

2. When it is determined that one RHR service water pump is inoperable, the redundant components of the

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
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 system are operable.	remaining subsystem shall be demonstrated to be operable immediately and daily thereafter.
3. From and after the date that one of the RHR service water systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the operable RHR service water system shall be demonstrated to be operable at least once each day.	3. When one RHR service water system becomes inoperable, the operable system shall be demonstrated to be operable immediately and daily thereafter.
4. To be considered operable, a RHR service water pump shall be capable of delivering 3500 gpm against a head of 500 feet.	
5. If the requirements of 3.5.C.1-3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION

High Pressure Core Cooling Capability

D. High Pressure Coolant Injection (HPCI) System

1. Except as specified in 3.5.D.2 below, the HPCI 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 HPCI system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all of the Automatic Pressure Relief system, the RCIC system, both of the core spray systems, and the LPCI subsystem and containment cooling mode of the RHR system are operable.

4.0 SURVEILLANCE REQUIREMENTS

High Pressure Core Cooling Capability

D. Surveillance of HPCI System shall be performed as follows:

1. Routine Testing

a. A simulated automatic actuation test shall be conducted each refueling outage.

b. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

2. When it is determined that HPCI system is inoperable, the RCIC system, the LPCI subsystem, and both of the core spray systems shall be demonstrated to be operable immediately.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. To be considered operable, the HPCI system shall meet the following conditions: a. The HPCI shall be capable of delivering 3,000 gpm into the reactor vessel for the reactor pressure range of 1120 psig to 150 psig. b. The condensate storage tanks shall contain at least 75,000 gallons of condensate water. c. The controls for automatic transfer of the HPCI pump suction from the condensate storage tank to the suppression chamber shall be operable. 4. If the requirements of 3.5.D.1-2 cannot be met, an orderly reactor shutdown shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.	

E. Automatic Pressure Relief System

1. Except as specified in 3.5.E.2 and 3.5.E.3 below, the entire automatic pressure relief system shall be operable at any time the reactor pressure is above 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the automatic pressure relief system valves is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days both remaining automatic relief system valves and the HPCI system are operable.

3. From and after the date that more than one of the automatic pressure relief valves are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI system is operable.

4. If the requirements of 3.5.E.1-3 cannot be met, an orderly reactor shutdown shall be initiated immediately and the reactor shall be reduced to 150 psig within 24 hours thereafter.

E. Surveillance of the Automatic Pressure Relief System shall be performed as follows:

1. Routine Testing

   a. A simulated automatic actuation test shall be conducted each operating cycle.

   b. Once each operating cycle, valve operability shall be verified by cycling the valves and observing a compensating change in turbine bypass valve position.

   c. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

2. **When it is determined that one or more automatic pressure relief valves of the Automatic Pressure Relief system is inoperable, the HPCI system shall be demonstrated to be operable immediately and weekly thereafter.**

3.0 LIMITING CONDITIONS FOR OPERATION

F. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in 3.5.F.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.

   a. To be considered operable, the RCIC system shall be capable of delivering 400 gpm into 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 15 days unless such system is sooner made operable, provided that during such 15 days all active components of the HPCI system are operable.

3. If the requirements of 3.5.F.1 - 2 cannot be met, an orderly shutdown of the reactor shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.

4.0 SURVEILLANCE REQUIREMENTS

F. Surveillance of Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Routine Testing

   a. A simulated automatic actuation test shall be conducted each refueling outage.

   b. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

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

G. 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 water temperature shall be reduced to less than 212°F 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.

G. Surveillance of Core and Containment Cooling System

1. When it is determined that one of the standby diesel generators is inoperable, all low pressure core cooling and containment cooling service water systems connected to the operable diesel generator 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.

3.0 LIMITING CONDITIONS FOR OPERATION

I. Recirculation System

1. Except as specified in 3.5.1.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

3. Valves in the equalizer piping between the recirculation loops shall be closed. Reactor operation with one loop shall be limited to 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

I. Recirculation System

1. Once per month, when irriated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cros tie valves cannot be opened using the normal control switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

3. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

Bases 4.5:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel, which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation will initially be functionally tested once per month until a trend is established and thereafter according to Figure 4.1 (see Section 3.1/4.1) with an interval not greater than three months. Core and containment cooling system components are inspected and tested in accordance with the requirements of 10 CFR 50, Section 50.55a(g). These requirements are delineated in Specification 4.13. This inspection and testing program, combined with the additional surveillance requirements contained in this section, provide a high degree of assurance that the core and containment cooling systems will perform as required when needed.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F :

   a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

E. Safety/Relief Valves

1. The integrity of the safety/relief valve bellows shall be continuously monitored.

2. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

3. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

F. deleted

G. Jet Pumps

Whenever the reactor is in the Startup or Run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the plant shall be placed in a cold shutdown condition within 24 hours.

F. deleted

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the Startup or Run modes, jet pump operability shall be checked daily by verifying that all the following conditions do not occur simultaneously:

1. The two recirculation loop flows are unbalanced by 15% or more when the recirculation pumps are operating at the same speed.

2. The indicated value of core flow rate is 10% or more less than the value derived from loop flow measurements.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
5. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next license amendment request.	The required inspection interval shall not be lengthened more than one step at a time. Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule. 2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days. 3. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up, and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers designated in Table 3.6.1 as being especially difficult to remove or located in High Radiation Areas during shutdown are exempt from this requirement. 4. Snubbers may be reclassified as being in or out of High Radiation Areas during shutdown in Table 3.6.1 based on the most recent radiation survey provided that a revision to Table 3.6.1 is included with the next license amendment request.

TABLE 3.6.1 (Page 1 of 2)
SAFETY RELATED SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
PS1-H2	MAIN STEAM	DRYWELL	953	071			I
PS1-H3	MAIN STEAM	DRYWELL	950	148			I
PS2-H2	MAIN STEAM	DRYWELL	950	120			I
PS3-H2	MAIN STEAM	DRYWELL	950	240			I
PS4-H3	MAIN STEAM	DRYWELL	950	212			I
RV24-H3	SAFETY-RELIEF	DRYWELL	950	110			I
RV24-H4	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H4A	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H5	SAFETY-RELIEF	DRYWELL	935	110			I
RV24A-H4A	SAFETY-RELIEF	DRYWELL	947	048		X	I
RV24A-H7	SAFETY-RELIEF	DRYWELL	953	115			I
RV24A-H8	SAFETY-RELIEF	DRYWELL	939	032			I
RV25-H1	SAFETY-RELIEF	DRYWELL	953	180			I
RV25-H1A	SAFETY-RELIEF	DRYWELL	953	180	X		I
RV25-H2	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H2A	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H3	SAFETY-RELIEF	DRYWELL	934	180	X		I
RV25A-H2	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H2A	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H7	SAFETY-RELIEF	DRYWELL	953	135			I
RV26-H1	SAFETY-RELIEF	DRYWELL	953	200	X		I
RV26-H1A	SAFETY-RELIEF	DRYWELL	953	200			I
RV26-H2	SAFETY-RELIEF	DRYWELL	947	200		X	I
RV26-H2A	SAFETY-RELIEF	DRYWELL	947	200			I
RV26A-H2	SAFETY-RELIEF	DRYWELL	940	250			I
RV26A-H2A	SAFETY-RELIEF	DRYWELL	935	250			I
RV27-H1	SAFETY-RELIEF	DRYWELL	950	320			I
RV27-H1A	SAFETY-RELIEF	DRYWELL	950	230			I
RV27-H5	SAFETY-RELIEF	DRYWELL	945	270			I
RV27-H6	SAFETY-RELIEF	DRYWELL	945	270			I
RV27A-H2A	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H3	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H9	SAFETY-RELIEF	DRYWELL	938	290			I
SS-1	MAIN STEAM	DRYWELL	953	279	X		I
SS-1AR	RECIRCULATION	DRYWELL	922	315	X	X	I
SS-1BR	RECIRCULATION	DRYWELL	922	135	X	X	I
SS-11	FEEDWATER	DRYWELL	952	302			I
SS-12	FEEDWATER	DRYWELL	952	058			I
SS-13	FEEDWATER	DRYWELL	952	258			I
SS-14	FEEDWATER	DRYWELL	952	096			I
SS-17A	RHR	DRYWELL	964	072	X		I
SS-17B	RHR	DRYWELL	964	072	X		I
SS-18A	RHR	DRYWELL	964	288			I
SS-18B	RHR	DRYWELL	964	288			I
SS-19	RHR	DRYWELL	964	341			I
SS-2	MAIN STEAM	DRYWELL	953	081	X		I
SS-2AR	RECIRCULATION	DRYWELL	927	302	X	X	I

TABLE 3.6.1 (Page 2 of 2)
SAFETY RELATED SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
SS-2BR	RECIRCULATION	DRYWELL	927	122		X	I
SS-20	RHR	DRYWELL	964	019	X		I
SS-3	MAIN STEAM	DRYWELL	950	212			I
SS-3AR	RECIRCULATION	DRYWELL	927	328		X	I
SS-3BR	RECIRCULATION	DRYWELL	927	148		X	I
SS-4	MAIN STEAM	DRYWELL	950	148			I
SS-4AR(A)	RECIRCULATION	DRYWELL	934	302			I
SS-4AR(B)	RECIRCULATION	DRYWELL	934	323			I
SS-4BR(A)	RECIRCULATION	DRYWELL	934	120			I
SS-4BR(B)	RECIRCULATION	DRYWELL	934	149			I
SS-40	HPCI	MAIN STEAM CHASE					I
SS-5AR	RECIRCULATION	DRYWELL	941	315		X	I
SS-5BR	RECIRCULATION	DRYWELL	941	135		X	I
SS-6AR	RECIRCULATION	DRYWELL	953	261	X		I
SS-6BR	RECIRCULATION	DRYWELL	953	099	X		I
SS-7	MAIN STEAM	DRYWELL	953	240	X		I
SS-7AR	RECIRCULATION	DRYWELL	953	323			I
SS-7BR	RECIRCULATION	DRYWELL	953	032			I
SS-8	MAIN STEAM	DRYWELL	953	120	X		I
SS-8AR	RECIRCULATION	DRYWELL	927	270		X	I
SS-8BR	RECIRCULATION	DRYWELL	927	090		X	I
SS-21	RHR	TORUS FL LV - S WALL					A
SS-22	RHR	TORUS FL LV - S WALL					A
SS-23	RHR	B RHR ROOM FL LV					A
SS-24	RHR	A RHR ROOM FL LV					A
SS-25	RHR	TORUS CATWK-SE WALL					A
SS-26	CORE SPRAY	B RHR ROOM FL LVL					A
SS-27	CORE SPRAY	B RHR ROOM FL LVL					A
SS-28A	CORE SPRAY	A RHR ROOM FL LVL					A
SS-28B	CORE SPRAY	A RHR ROOM FL LVL					A
SS-29	RHR	OVER N2 ANALYZER	954			X	A
SS-30	RHR	OVER N2 ANALYZER	954			X	A
SS-31	NHR	TORUS CATWK					A
SS-32A	RHR	A RHR ROOM - BY HX	916			x	A
SS-32B	RHR	A RHR ROOM - BY HX	916			X	A
SS-33	RHR	ABOVE TORUS					A
SS-34	RHR	ABOVE TORUS					A
SS-35	HPCI	HPCI ROOM - N WALL	912			x	A
SS-36A	HPCI	HPCI ROOM - FL LVL					A
SS-36B	HPCI	HPCI ROOM - FL LVL					A
SS-37	HPCI	HPCI ROOM - W WALL	905			x	A
SS-38A	RCIC	RCIC ROOM - W WALL	906			x	A
SS-38B	RCIC	RCIC ROOM - W WALL	906			x	A
SS-41	CORE SPRAY	ABOVE TORUS CATWK	927				A
SS-42	HPCI	ABOVE TORUS RING HDR	906				A

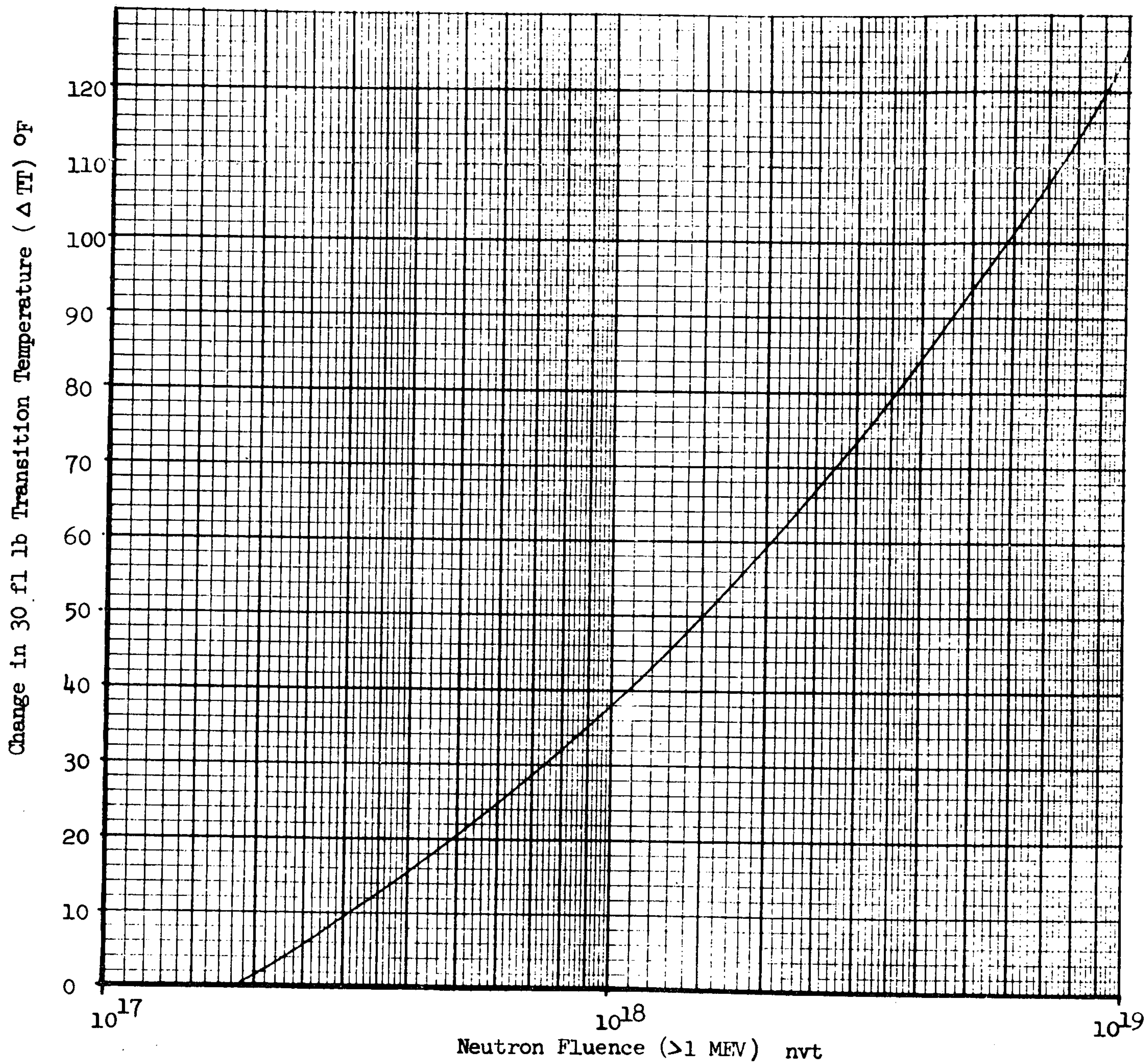


FIGURE 3.6.1 Change in Charpy V Transition Temperature versus Neutron Exposure

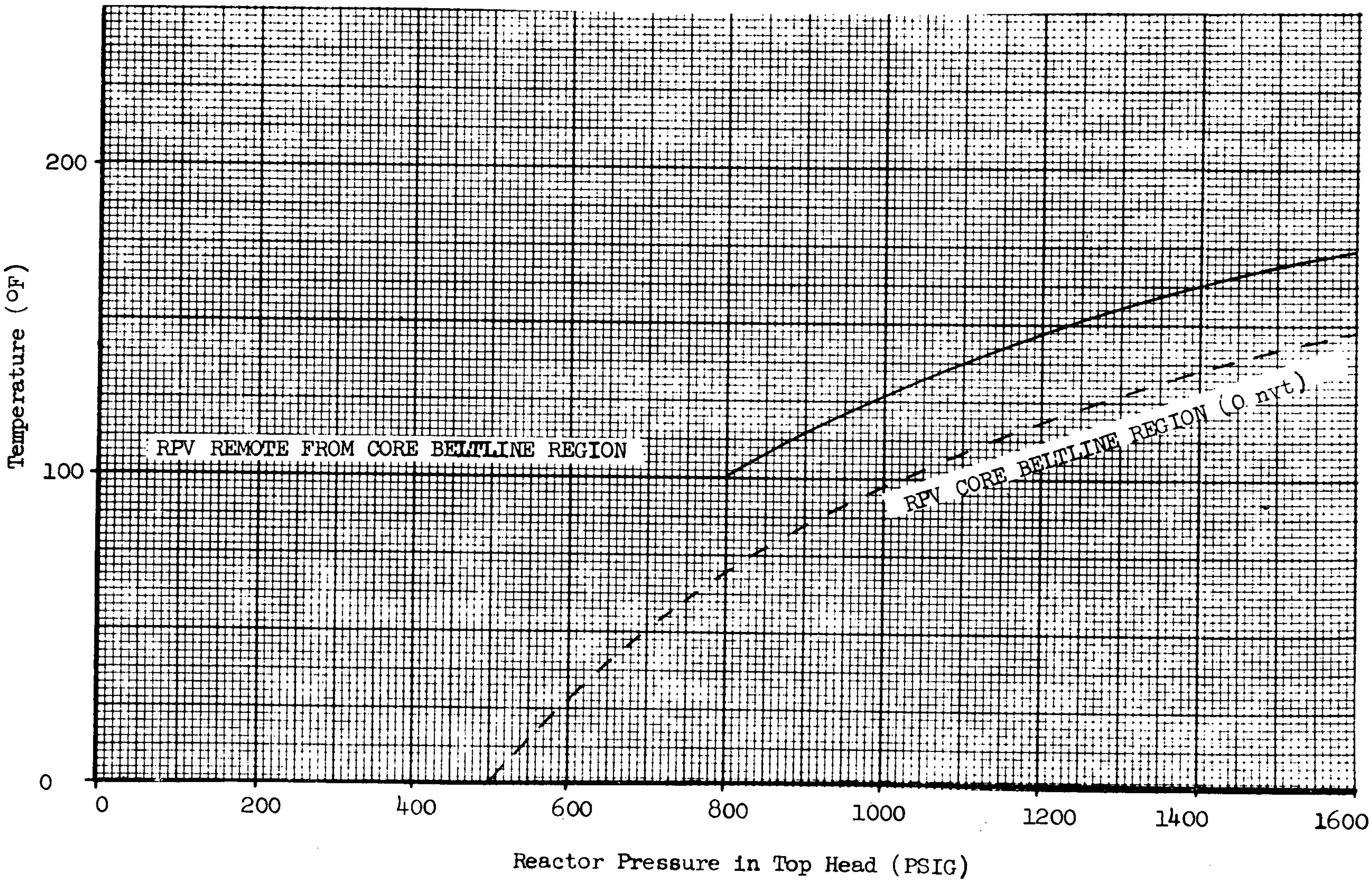


FIGURE 3.6.2 Minimum Temperature versus Pressure for Pressure Tests

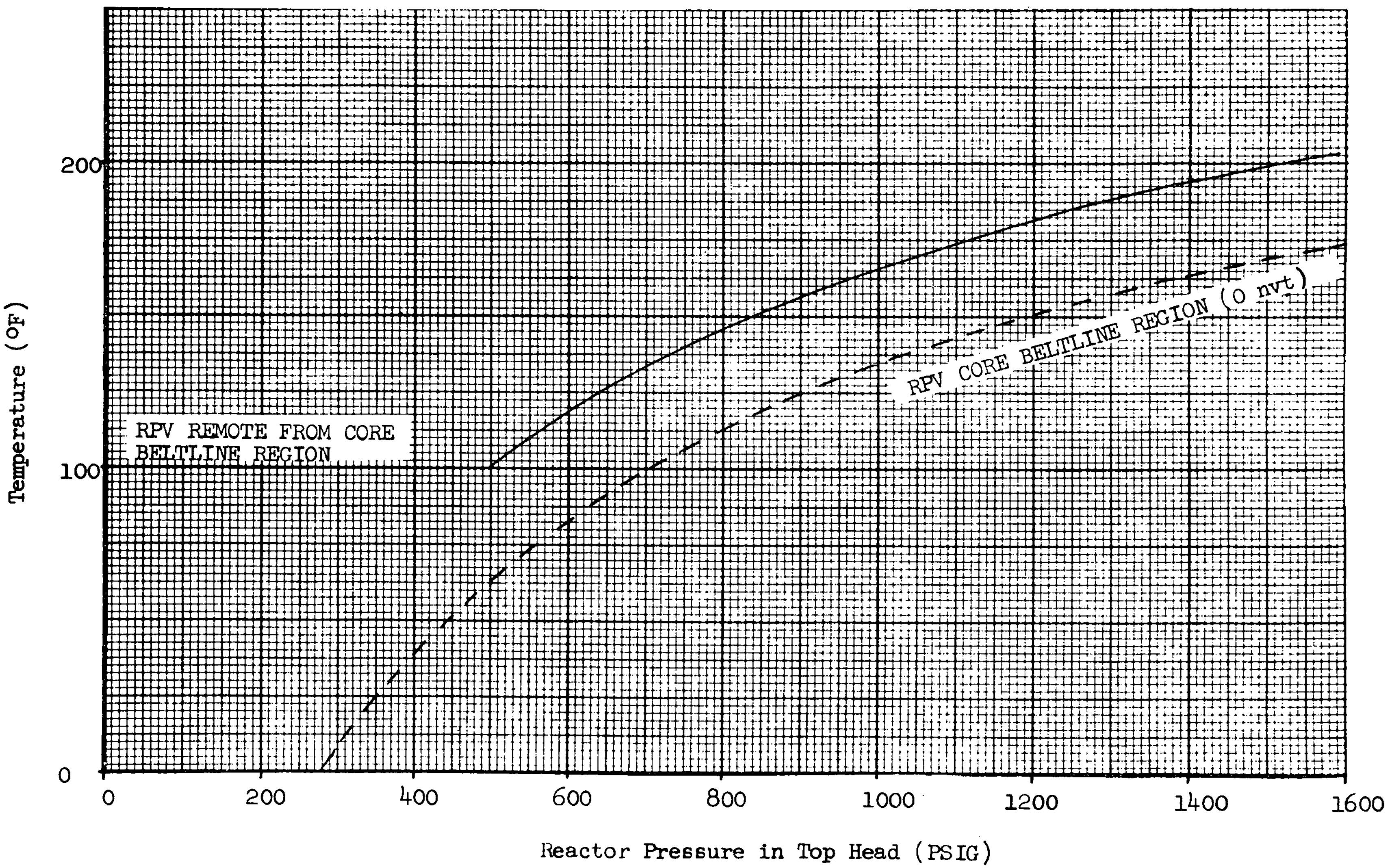


FIGURE 3.6.3 Minimum Temperature versus Pressure for Mechanical Heatup or Cooldown Following Nuclear Shutdown

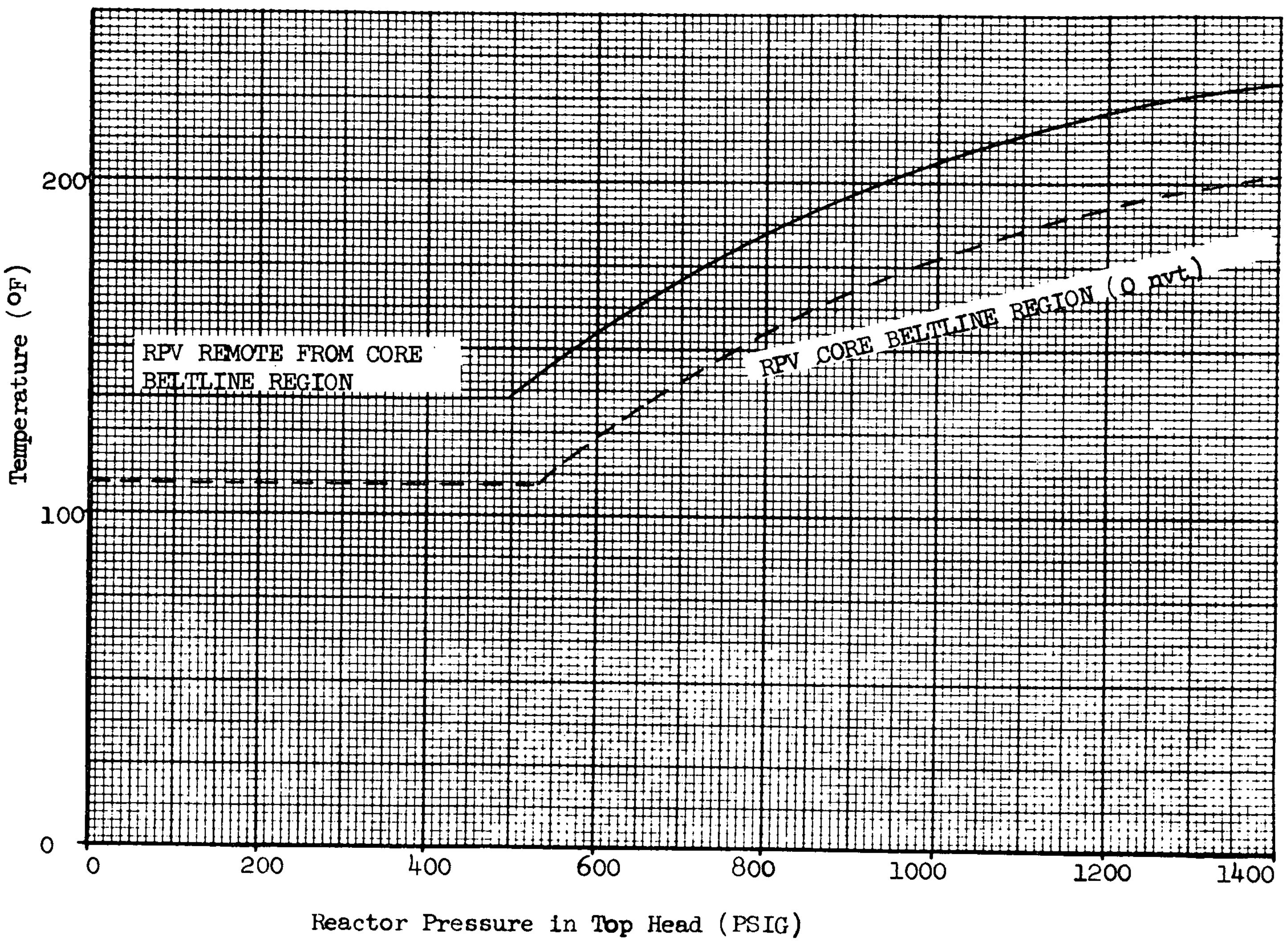
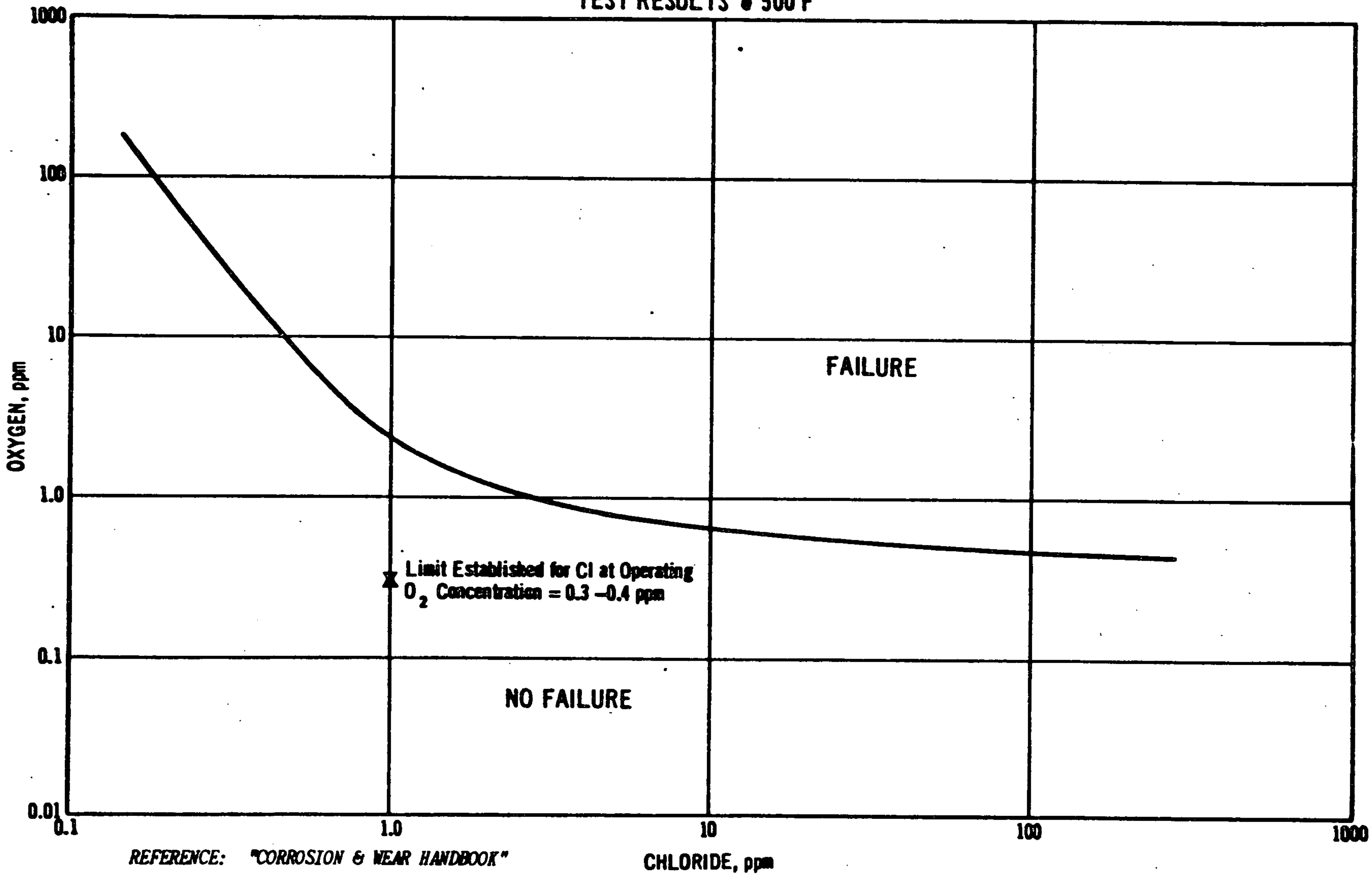


FIGURE 3.6.4 Minimum Temperature versus Pressure for Core Operation

FIGURE 4.6.2
CHLORIDE STRESS CORROSION
TEST RESULTS @ 500 F



REFERENCE: "CORROSION & WEAR HANDBOOK"
D. J. DePaul, Ed.

Bases Continued 3.6 and 4.6:

The safety/relief valves have two functions; i.e. power relief or self-actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

It is realized that there is no way to repair or replace the bellows during operation and the plant must be shut down to do this. The thirty-day period to do this allows the operator flexibility to choose his time for shutdown; meanwhile, because of the redundancy present in the design and the continuing monitoring of the integrity of the other valves, the overpressure pressure protection has not been compromised. The auto-relief function would not be impaired by a failure of the bellows. However, the self-actuated overpressure safety function would be impaired by such a failure.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provides assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

The program of safety/relief valve testing conforms to the requirements of 10 CFR 50, Section 50.55a(g). These requirements are delineated in Specification 4.13. This inspection and testing program, combined with the additional surveillance requirements contained in this section, provide a high degree of assurance that the safety/relief valves will perform as required when needed.

Bases Continued 3.6 and 4.6:

F. deleted

Bases Continued 3.6 and 4.6:

G. 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 occurred, 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 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 flow 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 through 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 measured. 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.

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.

Bases Continued 3.6 and 4.6:

H. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.H.4 prohibits startup with inoperable snubbers.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

3.0 LIMITING CONDITIONS FOR OPERATION

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers 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 vacuum breakers shall be 0.5 psi.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the suceeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4.0 SURVEILLANCE REQUIREMENTS

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

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

   b. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

d. The fuel cask or irradiated fuel is not being moved within the reactor building.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. Inservice inspection and testing of components shall be conducted in accordance with Specification 4.13.

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor operation in the run mode may continue provided at least one valve in each line having an inoperable valve is closed.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, initiate normal orderly shutdown and have reactor in the cold shutdown condition within 24 hours.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one fully closed valve in each line having an inoperable valve shall be recorded daily.

3.0 LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To assure the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. To be considered operable, Quality Group A, B, and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components, respectively, except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

4.13 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to the periodic inspection and testing of components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To verify the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. Inservice inspection of Quality Group A, B, and C components shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.0 LIMITING CONDITIONS FOR OPERATION

B. Inservice Testing of Pumps and Valves

1. To be considered operable, Quality Group A, B, and C pumps and valves shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for operability of ASME Code Class 1, 2, and 3 pumps and valves, respectively, except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

B. Inservice Testing of Pumps and Valves

1. Inservice testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been requested from the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Bases 3.13 and 4.13:

The inservice inspection and testing program conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection and testing of components classified into NRC Quality Groups A, B, and C will conform to the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code.

Using Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as a guide, all Monticello components have been classified into Quality Groups. This classification serves as the basis for determining which ASME Code Class inspection and testing requirements are applicable to a given component. 10 CFR 50, Section 50.55a(g) requires components which are part of the reactor coolant pressure boundary and their supports to meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 1. Other safety-related components must meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 2 or 3.

The inservice inspection program must be updated at 40 month intervals. The program for testing pumps and valves for operational readiness must be updated every 20 months. A description of the updated programs should be submitted to the NRC for review at least 90 days before the start of each period. A suggested format for this description is contained in Appendix A to reference (1).

The inservice inspection and testing program must, to the extent practical, comply with the requirements in editions and addenda to the ASME Code that are "in effect" no more than six months before the start of the period covered by the updated program. The term "in effect" means both having been published by the ASME, and having been referenced in paragraph (b) of 10 CFR 50, Section 50.55a. If a code required inspection or test is impractical, requests for deviations are submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i). The information specified in Appendix B to reference (1) should be submitted for each deviation requested. Deviation requests should, if possible, be submitted to the NRC for review at least 90 days before the start of each period. Deviations identified during an inspection period may be grouped and requested at the end of each calendar quarter. It is expected that a small number of deviations will be identified during the inspection period, particularily the first period when new inspection and testing techniques will be utilized. A requested deviation request may be considered acceptable to the Commission until a formal disapproval has been received.

References:

1. Letter from D. L. Ziemann, Chief, Operating Reactors Branch No. 2, USNRC, to L. O. Mayer, NSP, dated November 24, 1976.