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VAN BRUNT,E.E. Arizona Public Service Co.
RECIP.NAME RECIPIENT AFFILIATION
KNIGHTON,G. Licensing Branch 3

SUBJECT: Forwards response to 840814 request for review of proof & review Tech Specs.Function of committee for review & comment on Rev 3 to NUREG-0212 provided.Implementation of encl changes necessary for certification of Tech Specs.

SGB Tech Specs

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NOTES:Standardized plant. 05000528
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ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix, Arizona 85036

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced, units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

B001 Limited Dist

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt / FBK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc

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	J. B. Martin	(w/a)
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K. W. Gross
S. R. Frost
J. E. Kirby
D. R. Canady
W. F. Fernow
W. J. Jump
J. D. Houchen

PG 1-9 Add # note to startup Not accepted

me

JUSTIFICATION:

PVNGS Operations and engineering personnel needs this change to clarify and locate the special test exception for initial criticality and low power physics testing. Since this condition is not "Normal" Operating Procedure when the operations go critical below 350°F they may try to take some kind of corrective action to comply with Tech Specs not realizing that they are operating in a special test exception area. This change will assist Resident Inspectors identify and locate a Special Test Exception Factor.

PG 2-3 TABLE 2.2-1 ITEM 1.A.7.c RSB

Change allowable value to 42.1%.

JUSTIFICATION:

New CE number.

PG 2-5 ADD to FOOTNOTE 6 RSB

Setpoints are % of 100% power flow conditions.

JUSTIFICATION:

CE words to clarify and explain what exactly is happening.

PG B2-4 ADD TO STEAM GENERATOR LEVEL - LOW RSB

...10 minutes before auxiliary feedwater is required to prevent degraded core cooling

JUSTIFICATION:

PVNGS uses the term auxiliary vs emergency emergency as shown in the Tech Spec. Also the added phrase provides more detailed description as to what is really happening.

PG B2-5 CHANGE DNBR-LOW RSB

...Floor of 1861 psia.

JUSTIFICATION:

CE changed number

PG B2-6 CHANGE:

See page.

JUSTIFICATION:

CE changed numbers.
New descriptions and additions for Steam Generator Level High and Reactor Coolant Flow - Low provide more accurate and detailed information to assist the Operating Personnel.

RSB CPB

PG B2-7 CHANGE:

See page.

~~RSB~~ CPB

JUSTIFICATION:

Again, to provide a more accurate statement for CPC Addressable Constants.

PG 3/4 1-1 CHANGE ACTION STATEMENT

See page.

~~RSB~~ CPB

JUSTIFICATION:

This change has been continuously discussed with the NRC. As the spec is presently written there is alot of operator confusion as to if the operator immediately initiates boration or goes to Spec. 3.1.3.6 which allows the operator to have two hours to take corrective action Many of our Operations People have some confusion which will lead them to taking various corrective actions. The proof and review LCO 3.1.1.1 is in conflict with LCO 3.1.3.6. The action statement in LCO 3.1.1.1 requires immediate boration to reestablish 6% delta K/K shutdown margin, if violated. However, LCO 3.1.3.6 allows two (2) hours to recover rod position above the Transient Insertion Limit of Figure 3.1.3 or Figure 3.1.4. Since Transient Insertion Limits on these curves represents a value of available shutdown margin equivalent to 6% delta K/K, this means there are two different action requirements for exceeding the same limit. In actuality, the only verification of shutdown margin that is required to be performed during normal critical operation is the CEA position verification.

PROOF AND

PG 3/4 1-5 NEW CURVE COMING TO REPLACE PRESENT PROOF/REVIEW CURVE CPB

PG 3/4 1-8 CHANGE IN LCO AND ACTION STATEMENT RSB

3.1.2.2 ...Two of the following four... Add action D. Delete 210°F.

JUSTIFICATION:

PVNGS has an additional boron flow path to take credit for. It's a flow path from the refueling water tank via a high pressure safety injection pump to the RCS.

The 210°F for 6% ΔK/K shutdown margin needs to be deleted. The spec needs to state ...Borate to a shutdown margin... of 6% Δ K/K. Since we are a zero release plant boration equivalent to 6% ΔK/KK at 210°F when we are at 350°F would require more liquid waste to be processed and stored causing us other problems over the life of the plant since we are a zero release plant. Also CE has verified that 4% ΔK/K is appropriate shutdown margin therefore we have a 2% ΔK/K band of conservation.

PG 3/4 1-10 DELETE: RSB

210°F.

JUSTIFICATION:

See above.

1-12 new graph coming to replace present one RSB

PG 3/4 1-13 DELETE: RSB

210°F.

JUSTIFICATION:

See above.

PG 3/4 1-14 CHANGE ACTION STATEMENT 1.b.1 RSB

1-15

See attached.

JUSTIFICATION:

Complete reliability on the boronometer as a redundant method of borou dilution verification severely restricts plant operations. Independent collection and analysis of RCS samples is equally sound and would allow core alteration operations to continue in the event of a boronometer failure.

It should also be noted that the independent collection and analysis methods are implemented and utilized in may operating plants.

The * asterisk needs to be added to detail how various sampling will be conducted at various RCP operating conditions. This clarifies where these samples well be taken.

PG 3/4 1-17 ADD TO LCO - DELETE ACTION ITEM C CPB

JUSTIFICATION:

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors and the DNB power Operation Limit (PDL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPC's will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transient which is the limiting AOO for the PVNGS plants. When the COLSS is Out of Service (COOS) the monitoring function is performed via the CPC calculation of DNBR in conjunction with a Technical Specification COOS Limit Time which restricts the reactor power sufficiently to preserve the ROPM.

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNB and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event for which the penalty factor has been reduced is accounted for in two ways in the proposed change. The ROPM reserved in COLSS is used to account for some of the

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature	:	580°F
Low Pressure	:	1785 psia
ASI	:	-.3
Underflow fraction	:	0.865
Low Flow	:	95% of full flow
High Radial Peak	:	1.70 (Bank 5+4+PLR; PDIL = 40% Power; Reference 3)

Conclusion:

This revised Technical Specifications include a minimum power (50% of rated power) below which an additional power reduction is unnecessary, if there is a CEA misalignment with CEAC's out of service.

The minimum power for POL is 70%.

The added statement to action 6a is justified to the justification section for 3/4 1-17.

The work disabled in Items 6.b.1.c and 6.C.1.b are to be replaced with "Placed out of Service". This is consistent with PVNGS Terminology and also there is an indicator on the control panel that states "Out of Service" for the reactor power cutback system.

Delete the terms in 6.b.2.b and 6.C.2.b "...The inoperable status" to read ..."Be indicated that both CEACs are inoperable". This will avoid operator confusion and is consistent with how PVNGS Operators refer to this situation.

PG 3/4 3-9 CHANGE:

See Page.

RSB

ICSB

JUSTIFICATION:

CE number changes.

PG 3/4 3-10 DELETE:

Current # note add new one. See page.

RSB

ICSB

JUSTIFICATION:

The new pound note had been proposed to the NRC for the past 2 years. This change is consistent and reflects the way PVNGS will test the response times. This change is also agreed upon by CE.

"The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input."

The Proof and Review copy, # note reads, "Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown." This requires pertubation of the reactor plant and excludes the capability to utilize test equipment to verify response time. Pertubating the plant is an unnecessary safety and radiological (crud) risk.

PG 3/4 3-10a Type, see page. RSB ICSB

PG 3/4 3-12 <u>CHANGE</u>: RSB ICSB

III B to read change functional test R.

<u>JUSTIFICATION</u>:

Page 3/4 3-12, Table 4.3-1 III. B., Channel Functional Test column. Change to read: "R" (delete "M, S/U(1)").

This change is consistent with the SONGS 2 and SONGS 3 Tech Specs.

Table 4.3-1- III. A. provides adequate surveillance requirements for the Reactor Trip Breakers. The difference in plant equipment hardware between Table 4.3-1III. A. and III. B. are the reactor trip pushbutton switches. A surveillance test frequency of "R" is adequate for operability verification of pushbutton switches.

PG 3/4 3-13 <u>ADD</u> RSB ICSB

Statement, see page.

<u>JUSTIFICATION</u>:

Add the statement to note 8 for clarification and providing an alternate and more detailed method as to how to determine RCS flow rate.

PG 3/4 3-22, 3-23, Attachment 3-23-A <u>ADD</u> RSB ICSB

See page.

<u>JUSTIFICATION</u>:

Add the footnotes as indicated to provide detailed information to the plant staff as what actions to take to avoid confusion. The NR and WR identify if the readings are taken from the Wide Range or Narrow Range instrument.

PROOF AND ...

PG 3/4 3-24 Typo. accepted

~~ME~~ m2

PG 3/4 3-25 DELETE:

Old page. Insert new table.

RSB ICSB

JUSTIFICATION:

The new table is a more useable table to the operators. It breaks the systems down to where everyone understands. The old table was confusing in that the times represented showed the TOTAL time of the System response times not done of the key subsystem response times such as shown in the new table.

PG 3/4 3-26 DELETE:

Note. See page.

RSB ICSB

JUSTIFICATION:

The response times for the AFWP, are shown in the body of the table. Showing a note with another response time doesn't match is not really necessary. We have had alot of questions generated as to why the note if the AFWP, response times are in the Body of the table. Operations are continuously confused by both values; they want only 1 to worry about.

PG 3/4 3-27, 28, 29, 30, 31 Change automatic actuation logic channel functional test to R

RSB ICSB

JUSTIFICATION:

PVNGS will be submitted a letter to the NRC next week providing detailed justification. In summary we do not want to test this logic at power operation because various equipment would actuate and could trip the unit. Many other utilities are having problems meeting this Tech. Spec. and are going to be asking for a Tech. Spec change, in fact one set in there proposed change and justification about a month ago.

PG 3/4 3-34, 35 CHANGE:

Table 3.3-6. See page.

~~NAES~~ METB

JUSTIFICATION:

Incorporation of RETS ("Radiological Effluent Technical Specifications") into the STS ("Standard Technical Specifications") has resulted in duplication of operability requirements for the Effluent Monitoring System.

PROOF AND

MTEB

PG 3/4 3-64, 65, 66 67 **CHANGES:**

. Add footnotes.

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68 **DELETE:** ASB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is out belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-19 4-20 **CHANGE:** RSB

Tech. Spec. See page.

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 Typo. Mell

PG 3/4 4-29 NEW TABLE WILL BE SUPPLIED IN A WEEK RSB

✓ PG 3/4 4-9a Pressurizer Spray Valves New T/S 3.4.3.2 RSB

PG 3/4-34 RCS Vents New T/S 3.4.10

RSB

PG 3/4 4-32 **CHANGE:**

LCO and Action Item b. See page.

JUSTIFICATION:

CE Number Change.

RSB

PG 3/4 5-1 **CHANGE:**

RSB

LCO b. See Page.

review change to 2000 ppm.

JUSTIFICATION:

~~To clarify the Spec. for operations to avoid further confusion. Also~~ The 2000 ppm change is based on a new CE Number.

PG 3/4 5-5 **CHANGE:**

RSB

(a). See page.

JUSTIFICATION:

CE new number.

PG 3/4 5-6 **CHANGE:**

RSB

HPSI System - Single Pump. See page.

JUSTIFICATION:

This specification needs to be changed to comply with how we tested the pump during startup as well as how we will test the HPSI pump during operations. This change is in compliance with CE's design criteria.

Change LPSI Pump Sections 1 and 3 from leg to loop.

PG 3/4 6-1 **CHANGE:**

Surveillance 4.6.1.1a. Table 3.6-0. See pages. CSB

JUSTIFICATION:

Operations Department requests the attached changes to the PVNGS Technical Specification. Justification for each proposed change is summarized as follows:

Technical Specification 3/4.6.1.1. "Containment Integrity" and Technical Specification 3/4.6.3, "Containment Isolation Valves."

PG3/4 7-9a "Atmosphere Dump Valves
New T/S 3.7.1.6

RSB
ASB

ADD:

..., "sealed, or otherwise secured in at the open position." This statement was added to comply with the way PVNGS does business.

PG 3/4 7-6 CHANGE: ASB

Applicability: See page.

JUSTIFICATION:

The footnote was added to identify that when cooldown is in progress that the LCO in modes 3 and 4 do not require that the condensate storage tank does not require 300,000 gallons of water. This has been verified with our Safety Analysis.

PG 3/4 7-9 CHANGE: ASB

See page.

JUSTIFICATION

This input was supplied by the valve manufacturer.

PG 3/4 7-10 CHANGE: ASB

Surveillance. -

Change once per hour to once per shift.

JUSTIFICATION:

During our startup testing this spec required someone to take hourly data. It was noticed that there was not much change in temperature and pressure conditions when taken hourly and compared to an 8 hour shift.

PG 3/7 7-16 CHANGE: ~~AEB~~ ~~ASB~~ AEB

Surveillance Requirement 4.7.7.d.3.
... equal 1/8 inch ...

JUSTIFICATION:

This is the value Bechtel recommends. We did meet this during startup testing. This is also in compliance with the Reg Guide.

7-14 See page 33 ASB AEB

PG 3/4 7-39 CHANGE: CEB

Surveillance 4.7.11.6.b. See page.

JUSTIFICATION:

Delete ... "and verifying that the hydrant barrel is dry "... This Spec is used for those plants in climates where freezing occurs. PVNGS, as discussed in the FSAR, is not subject to climates or weather that would cause water in the hydrant barrel from freezing and causing various damage.

PG 3/4 7-41 CHANGE: CEB

LCO and Surveillance 4.7.12.1.b. See page.

JUSTIFICATION:

Delete the fire windows reference. PVNGS doesn't have fire windows.

PG 3/4 7-42 CHANGE: CEB

Surveillance 4.7.12.2. See page.

JUSTIFICATION:

Deletion of Item A is justified in that PVNGS doesn't have any fire door supervision system as described; therefore, we cannot comply with this.

PG 3/4 7-43 CHANGE: RSB

Surveillance 4.7.13.b. See page.

JUSTIFICATION:

CE new numbers.

PG 3/4 8-1 CHANGE:
8-2
8-7

Tech. Spec. See pages. RSB

JUSTIFICATION: ORAB

The Tech. Spec. was changed to comply with NRC Generic letter 84-15.

Technical Specification 3/4.8.1, "A.C. Sources"

The proposed changes to Tech. Spec. 3/4.8.1 are a result of the applicable recommendations of NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability" dated July 2, 1984.

PG 3/4 9-12 **DELETE:** me

See page.

JUSTIFICATION:

Typo.

PG 3/4 9-13 **CHANGE:** ASB AEB

LCO. See page.

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 **CHANGE:**

LCO Item C. See page. accepted ~~BSE~~ me

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 **CHANGE:** RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 **CHANGE:** ~~[illegible]~~ METB

Table 4.11-1. See page.

JUSTIFICATION:

Number left off Table.

PG 3/4 11-3 **CHANGE:** METB

Footnote b. See page.

5

PG 5-7 5-8 CHANGE:

RSB
CPB

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

CH6 CHANGE:

See next page

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

BASIS CHANGES

See next page

See pages.

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB	√ B3/4 7-1	RSB ASB
B 3/4 1-2	CPB	7-2	RSB ASB
1-3	CPB	7-3	ASB
2-1	CPB	7-4	MEB
3-3	CEB	7-6/7	CEB
3-4	ASB	9-3	AEB ASB
4-1	RSB	10-1	CPB
4-6	RSB	10-2	RSB
4-7	RSB	11-1/2	RAB METB
4-11	RSB	11-5	METB
5-1	CPB		
5-3	RSB		
6-2	CSB		
6-3/4	AEB CSB		

PG 3/4 2-5   CHANGE LCO and Surveillance - 4.2.4.4   LPB

JUSTIFICATION: Table 3.3-1 to be submitted later

Action 6 applies to this LCO not just 6C as stated. Also change the LCO to say ...Table 3.3-1. The rest of the sentence can be deleted because it is included in Action 6.

Change 4.2.4.4 to ...once per 31 EFPD because this surveillance refers to a burnup period vs. a calandar period. This change is in compliance with the way PVNGS intends to operate. If the unit is shutdown for any period of time and nothing was done to change the COLSS or CPC DNBR parameters/calculations they do not need to be verified.

PGS 3/4 2-~~8~~ 7   THESE GRAPHS WILL BE SUPPLIED IN ABOUT A WEEK   LPB
3/4 2-10

PGS 3/4 3-7,   CHANGE: RSB
3/4 3-8,
3/4 3-7a,   See attached.
3/4 3-8b

JUSTIFICATION:

Current Arizona Unit 1, Cycle 1 Technical Specifications does not specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEAC's out of service. An analysis was done to specify this lower power level.

This work is the completion of the CEAC's OOS work. This analysis improves ANPP Unit 1, Cycle 1 power capability from about 75% to greater than about 90% with both CEAC's out of service. The analysis of the documents and quality assures this result.

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEAC's out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

PROOF AND

MTEB

PG 3/4 3-64, 65, 66 67

CHANGES:

Add footnotes.

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68

DELETE:

ASB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is out belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-19 4-20

CHANGE:

RSB

Tech. Spec. See page.

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 Typo.

PG 3/4 4-29 NEW TABLE WILL BE SUPPLIED IN A WEEK

Mc

RSB

PG 3/4 4-9a Pressurizer Spray Valves New T/S 3.4.3.2

RSB

PG 3/4-34 RCS Vents New T/S 3.4.10 RSB

PG 3/4 4-32 **CHANGE:** RSB

LCO and Action Item b. See page.

JUSTIFICATION:

CE Number Change.

PG 3/4 5-1 **CHANGE:** RSB

LCO b. See Page. review change to 2000 ppm.

JUSTIFICATION:

~~To clarify the Spec. for operations to avoid further confusion. Also~~ The 2000 ppm change is based on a new CE Number.

PG 3/4 5-5 **CHANGE:** RSB

(a). See page.

JUSTIFICATION:

CE new number.

✓PG 3/4 5-6 **CHANGE:** RSB

HPSI System - Single Pump. See page.

JUSTIFICATION:

This specification needs to be changed to comply with how we tested the pump during startup as well as how we will test the HPSI pump during operations. This change is in compliance with CE's design criteria.

Change LPSI Pump Sections 1 and 3 from leg to loop.

PG 3/4 6-1 **CHANGE:**

Surveillance 4.6.1.1a. Table 3.6-0. See pages. CSB

JUSTIFICATION:

Operations Department requests the attached changes to the PVNGS Technical Specification. Justification for each proposed change is summarized as follows:

Technical Specification 3/4.6.1.1. "Containment Integrity" and Technical Specification 3/4.6.3, "Containment Isolation Valves."

PG3/4 7-9a "Atmosphere Dump Valves New T/S 3.7.1.6 ~~BSB~~ ASB

ADD:

..., "sealed, or otherwise secured in at the open position." This statement was added to comply with the way PVNGS does business.

PG 3/4 7-6 CHANGE: ASB

Applicability: See page.

JUSTIFICATION:

The footnote was added to identify that when cooldown is in progress that the LCO in modes 3 and 4 do not require that the condensate storage tank does not require 300,000 gallons of water. This has been verified with our Safety Analysis.

PG 3/4 7-9 CHANGE: ASB

See page.

JUSTIFICATION

This input was supplied by the valve manufacturer.

PG 3/4 7-10 CHANGE: ASB

Surveillance.

Change once per hour to once per shift.

JUSTIFICATION:

During our startup testing this spec required someone to take hourly data. It was noticed that there was not much change in temperature and pressure conditions when taken hourly and compared to an 8 hour shift.

PG 3/7 7-16 CHANGE: ~~AEB~~ ~~BHB~~ AEB

Surveillance Requirement 4.7.7.d.3.
... equal 1/8 inch ...

JUSTIFICATION:

This is the value Bechtel recommends. We did meet this during startup testing. This is also in compliance with the Reg Guide.

7-14 See page 33 ASB AEB

PG 3/4 7-39 CHANGE: CEB

Surveillance 4.7.11.6.b. See page.

JUSTIFICATION:

Delete ... "and verifying that the hydrant barrel is dry "... This Spec is used for those plants in climates where freezing occurs. PVNGS, as discussed in the FSAR, is not subject to climates or weather that would cause water in the hydrant barrel from freezing and causing various damage.

PG 3/4 7-41 CHANGE: CEB

LCO and Surveillance 4.7.12.1.b. See page.

JUSTIFICATION:

Delete the fire windows reference. PVNGS doesn't have fire windows.

PG 3/4 7-42 CHANGE: CEB

Surveillance 4.7.12.2. See page.

JUSTIFICATION:

Deletion of Item A is justified in that PVNGS doesn't have any fire door supervision system as described; therefore, we cannot comply with this.

PG 3/4 7-43 CHANGE: RSB

Surveillance 4.7.13.b. See page.

JUSTIFICATION:

CE new numbers.

PG 3/4 8-1 CHANGE:
8-2
8-7

Tech. Spec. See pages. PSB ORAB

JUSTIFICATION:

The Tech. Spec. was changed to comply with NRC Generic letter 84-15.

Technical Specification 3/4.8.1, "A.C. Sources"

The proposed changes to Tech. Spec. 3/4.8.1 are a result of the applicable recommendations of NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability" dated July 2, 1984.

PG 3/4 9-12 DELETE:

See page.

JUSTIFICATION:

Typo.

me

PG 3/4 9-13 CHANGE:

LCO. See page.

ASB AEB

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 CHANGE:

LCO Item C. See page.

accepted ~~RSB~~ me

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 CHANGE:

RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 CHANGE:

Table 4.11-1. See page.

~~PSSES~~ METB

JUSTIFICATION:

Number left off Table.

PG 3/4 11-3 CHANGE:

METB

Footnote b. See page.

PG 5-7 5-8 CHANGE: (RSB) CPB

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

CH6 CHANGE: See next page

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

BASIS CHANGES See next page

See pages.

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB	B3/4 7-1	RSB ASB
B3/4 1-2	CPB	7-2	RSB ASB
1-3	CPB	7-3	ASB
2-1	CPB	7-4	MEB
3-3	CEB	7-6/7	CEB
3-4	ASB	9-3	AEB ASB
4-1	RSB	10-1	CPB
4-6	RSB	10-2	RSB
4-7	RSB	11-1/2	~~RAB~~ RSB METB
4-11	RSB	11-5	METB
5-1	CPB		
5-3	RSB		
6-2	CSB		
6-3/4	AEB CSB		

PROOF AND REVIEW

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 2383 psia	≤ 2388 psia
2. Pressurizer Pressure - Low	≥ 1837 psia (2)	≥ 1822 psia (2)
3. Steam Generator Level - Low	≥ 44.2% (4)	≥ 43.7% (4)
4. Steam Generator Level - High	≤ 91.0% (9)	≤ 91.5% (9)
5. Steam Generator Pressure - Low	≥ 919 psia (3)	≥ 912 psia (3)
6. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
7. Reactor Coolant Flow - Low		
a. Rate	≤ 1.05%/s (6)(7)	≤ 1.10%/s (6)(7)
b. Floor	≥ 52.2% (6)(7)	≥ 47.2% (6)(7)
c. Band	≤ 40.0% (6)(7)	≤ ~~42.2%~~ (6)(7) 42.1%
8. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
9. DNBR - Low	≥ 1.231 (5)	≥ 1.231 (5)
B. Excore Neutron Flux		
1. Variable Overpower Trip		
a. Rate	≤ 10.6%/min of RATED THERMAL POWER (8)	≤ 11.0%/min of RATED THERMAL POWER (8)
b. Ceiling	≤ 110.0% of RATED THERMAL POWER (8)	≤ 111.0% of RATED THERMAL POWER (8)
c. Band	≤ 9.8% of RATED THERMAL POWER (8)	≤ 10.0% of RATED THERMAL POWER (8)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	≤ 0.798% of RATED THERMAL POWER	≤ 0.895% of RATED THERMAL POWER
b. Shutdown	≤ 0.798% of RATED THERMAL POWER	≤ 0.895% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	≤ ~~2434~~ 2409 psia	≤ ~~2439~~ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(6) <u>RATE</u> is the maximum rate of decrease of the trip setpoint.
<u>FLOOR</u> is the minimum value of the trip setpoint.
<u>BAND</u> is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor. SET POINTS ARE % OF 100% POWER FLOW CONDITIONS

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) <u>RATE</u> is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
<u>CEILING</u> is the maximum value of the trip setpoint.
<u>BAND</u> is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

PROOF AND REVIEW

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	≤ 0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.02
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before ~~emergency~~ AUXILIARY feedwater is required TO PREVENT DEGRADED CORE COOLING.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

PROOF AND REVIEW

Local Power Density - High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of ~~1785~~ 1861 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core

PROOF AND REVIEW

DNBR - Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	$\geq$ 470°F
b.	RCS Cold Leg Temperature-High	$\leq$ 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	$\geq$ ~~1860~~ psia 1861
f.	Pressurizer Pressure-High	$\leq$ ~~2339~~ psia 2388
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level - High

~~The Steam Generator Level - High trip provides protection in the event of excess feedwater flow. The setpoint for the trip is identical to the main steam isolation setpoint.~~

Reactor Coolant Flow - Low

A FOUR PUMP FLOW COASTDOWN DURING A STEAMLINE BREAK WITH A LOSS OF OFFSITE POWER

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions.

THE STEAM GENERATOR LEVEL-HIGH TRIP IS PROVIDED TO PROTECT THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. SINCE THE TURBINE IS AUTOMATICALLY TRIPPED WHEN THE REACTOR IS TRIPPED, THIS TRIP PROVIDES A RELIABLE MEANS FOR PROVIDING PROTECTION TO THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. THIS TRIP'S SETPOINT DOES NOT CORRESPOND TO A SAFETY LIMIT AND NO CREDIT WAS TAKEN IN THE ACCIDENT ANALYSES FOR OPERATION OF THIS TRIP. ITS FUNCTIONAL CAPABILITY AT THE SPECIFIED TRIP SETTING ENHANCES THE OVERALL RELIABILITY OF THE REACTOR PROTECTION SYSTEM

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following ~~three~~ FOUR boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

D. A FLOW PATH FROM THE REFUELING WATER TANK VIA A HIGH PRESSURE SAFETY INJECTION PUMP TO THE REACTOR COOLANT SYSTEM

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm to the Reactor Coolant System.

PROOF AND REVIEW

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

PROOF AND REVIEW

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

DATE

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

3/4 1-12

PROOF AND REVIEW

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F; restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

PROOF AND REVIEW

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Table 3.1-1 by either boronometer or RCS sampling.**

b. With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by both boronometer and RCS sampling when entering MODE 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 - 3.1-5, as applicable, by both boronometer and RCS sampling. If one of the methods of determining the RCS boron concentration is not available, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one additional method for detecting a boron dilution is restored to OPERABLE status.

See attached attachment 3/4 1-14A

2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK:

1. At least once per 12 hours.

2. When initially setting setpoints at the following times:

a) One hour after a reactor trip.

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

PROOF AND REVIEW

ATTACHMENT 3/4 1-14 A

With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by ~~either~~ either both boronmeter and RCS sampling* ~~or~~ by independent collection and analysis of two RCS samples when entering Mode 3,4,or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable montioring frequency in Tables 3.1-1 through 3.1.5, as applicable, by either boronmeter and RCS sampling*or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the methods for determining and confirming RCS boron concentration is restored.

SURVEILLANCE REQUIREMENTS (Continued)

b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

** WITH ONE OR MORE REACTOR COOLANT PUMPS (RCP) OPERATING. THE SAMPLE SHOULD BE OBTAINED FROM THE HOT LEG. WITH NO RCP OPERATING, THE SAMPLE SHOULD BE OBTAINED FROM THE DISCHARGE LINE OF THE LOW PRESSURE SAFETY INJECTION (LPSI) PUMP OPERATING IN THE SHUTDOWN COOLING MODE.

TABLE 3.1-1

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} > 0.98

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	1 hour	Operation not allowed	
4	12 hours	1 hour	Operation not allowed	
5 RCS filled	8 hours	1 hour	Operation not allowed	
5 RCS partially drained	Operation not allowed			
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.98 \geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	2.5 hours	1 hour	0.5 hours
4	12 hours	2.5 hours	1 hour	0.5 hours
5 RCS filled	8 hours	2.5 hours	1 hour	0.5 hours
5 RCS partially drained	8 hours	0.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

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OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps			
	0	1	2	3
3	12 hours	3.5 hours	1.5 hours	1 hour
4	12 hours	3.5 hours	1.5 hours	1 hour
5 RCS filled	8 hours	3.5 hours	1.5 hours	1 hour
5 RCS partially drained	8 hours	1 hour	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

TABLE 3.1-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	5 hours	2 hours	1 hour
4	12 hours	5 hours	2 hours	1 hour
5 RCS filled	8 hours	5 hours	2 hours	1 hour
5 RCS partially drained	8 hours	1.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-5

REQUIRED MONITORING FREQUENCIES ~~FOR BACKUP BORON DILUTION~~
DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS
AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	6 hours	3 hours	1.5 hours
4	12 hours	6 hours	3 hours	1.5 hours
5 RCS filled	8 hours	6 hours	3 hours	1.5 hours
5 RCS partially drained	8 hours	2 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restart periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	2#, 3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	2#, 3#
9. DNBR - Low	4	2 (c)(d)	3	1, 2	2#, 3#
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	2#, 3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	8
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1, 2	2#, 3#, 7

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
D. Supplementary Protection System					
Pressurizer Pressure - High	4 (f)	2	4	1, 2	8
II. RPS LOGIC					
A. Matrix Logic	6	1	3	1, 2	1
	6	1	3	3*, 4*, 5*	8
B. Initiation Logic	4	2	4	1, 2	5
	4	2	4	3*, 4*, 5*	8
III. RPS ACTUATION DEVICES					
A. Reactor Trip Breaker	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8
B. Manual Trip	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1. The trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. AFTER 7 DAYS, OPERATION MAY CONTINUE PROVIDED THAT THE CONDITIONS OF ACTION ITEM 6.b OR 6.c ARE MET

b. With both CEACs inoperable and COLSS in ~~operation~~, SERVICE operation may continue provided that:

1. Within 1 hour:

~~a)~~ The margins required by Specification 3.2.4 are increased and maintained at a value equivalent to or greater than the percentage of RATED THERMAL POWER shown on Figure 3.3-1.

c) The Reactor Power Cutback System is ~~disabled~~. PLACED OUT OF SERVICE

A) OPERATION IS RESTRICTED TO THE LIMITS SHOWN IN FIG 3.3-1. THE DNBR MARGIN REQUIRED BY SPECIFICATION 3.2.4 IS REPLACED BY THIS RESTRICTION WHEN BOTH CEAC'S ARE INOPERABLE AND COLSS IS IN OPERATION

b) THE LINEAR HEAT RATE MARGIN REQUIRED BY SPECIFICATION 3.2.1 IS MAINTAINED

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 2383 psia	≤ 2388 psia
2. Pressurizer Pressure - Low	≥ 1837 psia (2)	≥ 1822 psia (2)
3. Steam Generator Level - Low	≥ 44.2% (4)	≥ 43.7% (4)
4. Steam Generator Level - High	≤ 91.0% (9)	≤ 91.5% (9)
5. Steam Generator Pressure - Low	≥ 919 psia (3)	≥ 912 psia (3)
6. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
7. Reactor Coolant Flow - Low		
a. Rate	≤ 1.05%/s (6)(7)	≤ 1.10%/s (6)(7)
b. Floor	≥ 52.2% (6)(7)	≥ 47.2% (6)(7)
c. Band	≤ 40.0% (6)(7)	≤ ~~42.2%~~ (6)(7) 42.1%
8. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
9. DNBR - Low	≥ 1.231 (5)	≥ 1.231 (5)
B. Excore Neutron Flux		
1. Variable Overpower Trip		
a. Rate	≤ 10.6%/min of RATED THERMAL POWER (8)	≤ 11.0%/min of RATED THERMAL POWER (8)
b. Ceiling	≤ 110.0% of RATED THERMAL POWER (8)	≤ 111.0% of RATED THERMAL POWER (8)
c. Band	≤ 9.8% of RATED THERMAL POWER (8)	≤ 10.0% of RATED THERMAL POWER (8)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	≤ 0.798% of RATED THERMAL POWER	≤ 0.895% of RATED THERMAL POWER
b. Shutdown	≤ 0.798% of RATED THERMAL POWER	≤ 0.895% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	≤ ~~2434~~ 2409 psia	≤ ~~2439~~ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

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TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(6) RATE is the maximum rate of decrease of the trip setpoint.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor. SET POINTS ARE % OF 100% POWER FLOW CONDITIONS

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

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TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$ 1.15
61	FC2	Core coolant mass flow rate calibration constant	$\leq$ 0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	$\geq$ 1.02
64	TPC	Thermal power calibration constant	$\geq$ 0.90
65	KCAL	Neutron flux power calibration constant	$\geq$ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

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Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before ~~emergency~~ AUXILIARY feedwater is required TO PREVENT DEGRADED CORE COOLING.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

Local Power Density - High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of ~~1785~~ 1861 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core

DNBR - Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	$\geq$ 470°F
b.	RCS Cold Leg Temperature-High	$\leq$ 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	$\geq$ ~~1860~~ psia 1861
f.	Pressurizer Pressure-High	$\leq$ ~~2339~~ psia 2388
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level - High

~~The Steam Generator Level - High trip provides protection in the event of excess feedwater flow. The setpoint for the trip is identical to the main steam isolation setpoint.~~

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions.

A FOUR PUMP FLOW COASTDOWN DURING A STEAMLINE BREAK WITH A LOSS OF OFFSITE POWER

THE STEAM GENERATOR LEVEL-HIGH TRIP IS PROVIDED TO PROTECT THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. SINCE THE TURBINE IS AUTOMATICALLY TRIPPED WHEN THE REACTOR IS TRIPPED, THIS TRIP PROVIDES A RELIABLE MEANS FOR PROVIDING PROTECTION TO THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. THIS TRIP'S SETPOINT DOES NOT CORRESPOND TO A SAFETY LIMIT AND NO CREDIT WAS TAKEN IN THE ACCIDENT ANALYSES FOR OPERATION OF THIS TRIP. ITS FUNCTIONAL CAPABILITY AT THE SPECIFIED TRIP SETTING ENHANCES THE OVERALL RELIABILITY OF THE REACTOR PROTECTION SYSTEM

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REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following ~~three~~ FOUR boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

D. A FLOW PATH FROM THE REFUELING WATER TANK VIA A HIGH PRESSURE SAFETY INJECTION PUMP TO THE REACTOR COOLANT System

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm to the Reactor Coolant System.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

DATE

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F; restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4...2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

   1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Table 3.1-1 by either boronometer or RCS sampling.**

b. With both startup channel high neutron flux alarms inoperable:

   1. Determine the RCS boron concentration by both boronometer and RCS sampling when entering MODE 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 - 3.1-5, as applicable, by both boronometer and RCS sampling. If one of the methods of determining the RCS boron concentration is not available, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one additional method for detecting a boron dilution is restored to OPERABLE status.

   SEE ATTACHED ATTACHMENT 3/4 1-14A

   2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK:

   1. At least once per 12 hours.

   2. When initially setting setpoints at the following times:

      a) One hour after a reactor trip.

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

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ATTACHMENT 3/4 1-14 A

With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by ~~either~~ ~~either~~ both boronmeter and RCS sampling* or by independent collection and analysis of two RCS samples when entering Mode 3,4,or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable montioring frequency in Tables 3.1-1 through 3.1.5, as applicable, by either boronmeter and RCS sampling* or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the methods for determining and confirming RCS boron concentration is restored.

SURVEILLANCE REQUIREMENTS (Continued)

b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

** WITH ONE OR MORE REACTOR COOLANT PUMPS (RCP) OPERATING: THE SAMPLE SHOULD BE OBTAINED FROM THE HOT LEG. WITH NO RCP OPERATING, THE SAMPLE SHOULD BE OBTAINED FROM THE DISCHARGE LINE OF THE LOW PRESSURE SAFETY INJECTION (LPSI) PUMP OPERATING IN THE SHUTDOWN COOLING MODE.

TABLE 3.1-1

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} > 0.98$

OPERATIONAL MODE	Number of OPERATING ~~OPERABLE~~ Charging Pumps 0	1	2	3
3	12 hours	1 hour	Operation not allowed	
4	12 hours	1 hour	Operation not allowed	
5 RCS filled	8 hours	1 hour	Operation not allowed	
5 RCS partially drained	Operation not allowed			
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR 0.98 $\geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	2.5 hours	1 hour	0.5 hours
4	12 hours	2.5 hours	1 hour	0.5 hours
5 RCS filled	8 hours	2.5 hours	1 hour	0.5 hours
5 RCS partially drained	8 hours	0.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

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TABLE 3.1-3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	3.5 hours	1.5 hours	1 hour
4	12 hours	3.5 hours	1.5 hours	1 hour
5 RCS filled	8 hours	3.5 hours	1.5 hours	1 hour
5 RCS partially drained	8 hours	1 hour	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	5 hours	2 hours	1 hour
4	12 hours	5 hours	2 hours	1 hour
5 RCS filled	8 hours	5 hours	2 hours	1 hour
5 RCS partially drained	8 hours	1.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of ~~OPERABLE~~ OPERATING Charging Pumps 0	1	2	3
3	12 hours	6 hours	3 hours	1.5 hours
4	12 hours	6 hours	3 hours	1.5 hours
5 RCS filled	8 hours	6 hours	3 hours	1.5 hours
5 RCS partially drained	8 hours	2 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

SURVEILLANCE REQUIREMENTS (Continued)

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restart periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	$2^{\#}$, $3^{\#}$
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	$2^{\#}$, $3^{\#}$
6. Containment Pressure - High	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
9. DNBR - Low	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
	4	2	3	3*, 4*, 5*	8
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$, 7

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
D. Supplementary Protection System					
Pressurizer Pressure - High	4 (f)	2	4	1, 2	8
II. RPS LOGIC					
A. Matrix Logic	6	1	3	1, 2	1
	6	1	3	3*, 4*, 5*	8
B. Initiation Logic	4	2	4	1, 2	5
	4	2	4	3*, 4*, 5*	8
III. RPS ACTUATION DEVICES					
A. Reactor Trip Breaker	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8
B. Manual Trip	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1. The trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. AFTER 7 DAYS, OPERATION MAY CONTINUE PROVIDED THAT THE CONDITIONS OF ACTION ITEM 6.b OR 6.c ARE MET

b. With both CEACs inoperable and COLSS in ~~operation~~, SERVICE operation may continue provided that:

1. Within 1 hour:

~~a)~~ The margins required by Specification 3.2.4 are increased and maintained at a value equivalent to or greater than the percentage of RATED THERMAL POWER shown on Figure 3.3-1.

c) The Reactor Power Cutback System is ~~disabled~~. PLACED OUT OF SERVICE

A) OPERATION IS RESTRICTED TO THE LIMITS SHOWN IN FIG 3.3-1. THE DNBR MARGIN REQUIRED BY SPECIFICATION 3.2.4 IS REPLACED BY THIS RESTRICTION WHEN BOTH CEAC'S ARE INOPERABLE AND COLSS IS IN OPERATION

b) THE LINEAR HEAT RATE MARGIN REQUIRED BY SPECIFICATION 3.2.1 IS MAINTAINED

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status.~~ BE INDICATE THAT BOTH CEACS ARE INOPERABLE

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. FOLLOWING A CEA MISALIGNMENT, WITH BOTH CEAC'S INOPERABLE AND COLSS IN OPERATION, OPERATION MAY CONTINUE PROVIDED THAT:

   WITHIN 1 HOUR:

   A) THE POWER IS REDUCED TO 80% OF THE PRE-MISALIGNED POWER BUT NEED NOT BE REDUCED TO LESS THAN 60% OF RATED POWER.

   b) REFER TO SECTION 3.1.3, MOVABLE CONTROL ASSEMBLIES, FOR FURTHER SPECIFICATIONS ON CEA MISALIGNMENTS.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

1. Within 1 hour:

   a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by ~~1.18~~ 1.19 ~~and the~~ resulting value is re-entered into the CPCs.

   b) The Reactor Power Cutback System is ~~disabled~~. PLACED OUT OF SERVICE

   c) THE COLSS OUT OF SERVICE LIMIT LINE, SECTION 3.2.4 IS NOT APPLICABLE TO THIS MODE OF OPERATION.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status.~~ BE INDICATED THAT BOTH CEAC'S ARE INOPERABLE

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. FOLLOWING A CEA MISALIGNMENT, WITH BOTH CEAC'S AND COLSS INOPERABLE, OPERATION MAY CONTINUE PROVIDED THAT: WITHIN 1 HOUR:
a) THE POWER IS REDUCED TO 80% OF THE PRE-MISALIGNED POWER, BUT NEED NOT BE REDUCED TO LESS THAN 50% OF RATED POWER.
b) REFER TO SECTION 3.1.3, MOVABLE CONTROL ASSEMBLIES, FOR FURTHER SPECIFICATIONS ON CEA MISALIGNMENT

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - Wi . the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour.

DATE SUBJECT

DATE

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

DATE

JOB NO.

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 1.15 seconds
2. Pressurizer Pressure - Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ ~~0.75~~ second 0.65
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. Cold Leg Temperature	≤ ~~0.81~~ second## 0.75
d. Hot Leg Temperature	≤ ~~0.81~~ second## 0.75
e. Primary Coolant Pump Shaft Speed	≤ ~~0.52~~ second# 0.75
f. Reactor Coolant Pressure from Pressurizer	≤ ~~0.48~~ second### 0.75
g. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 1.15 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE [illegible] (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

[#Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown.]

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 13 seconds. Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accomodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

THE PULSE TRANSMITTERS MEASURING PUMP SPEED ARE EXEMPT FROM RESPONSE TIME TESTING. THE RESPONSE TIME SHALL BE MEASURED FROM THE PULSE SHAPE INPUT

PROOF AND REVIEW

TABLE 3.3-2a

INCREASES IN BERRO, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME (τ)	BERRO INCREASE (%)	BERR2 INCREASE (%)	BERR4 INCREASE (%)
$\tau \leq 8.0$ sec 0	0	0	0
8.0 sec $< \tau \leq$ 10.0 sec	2.5	2.0	1.0
10.0 sec $< \tau \leq$ 13.0 sec	6.0	4.0	6.0

NOTE: BERR term increases are not cumulative. For example, if the time constant changes from the range of $8.0 < \tau \leq 10.0$ sec to the range $10.0 < \tau \leq 13.0$, the BERRO increase from its original ($\tau \leq 8.0$ sec) value is 6.0 not 2.5 + 6.0.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	D (2, 4), R (4, 5)	M, R (6)	1, 2
9. DNBR - Low	S	D (2, 4), R (4, 5) M (8), S (7)	M, R (6)	1, 2
B. Excore Neutron Flux				
1. Variable Overpower Trip	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5 and *
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M, S/U (1) R	1, 2, 3*, 4*, 5*

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust he CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

EITHER USING THE REACTOR COOLANT PUMP DIFFERENTIAL PRESSURE INSTRUMENTATION AND THE ULTRASONIC FLOW METER ADJUSTED PUMP CURVES (CONSERVATIVELY COMPENSATED FOR MEASUREMENT UNCERTAINTIES) OR

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. SAFETY INJECTION (SIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a), 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
II. CONTAINMENT ISOLATION (CIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
II. CONTAINMENT ISOLATION (Continued)					
3. Manual CIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
4. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
III. CONTAINMENT SPRAY (CSAS)					
A. Sensor/Trip Units					
Containment Pressure -- High - High	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual CSAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
IV. MAIN STEAM LINE ISOLATION (MSIS)					
A. Sensor/Trip Units					
1. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(b), 4	13*, 14*
2. Steam Generator Level - High	4/steam generator	2/steam generator	3/steam generator	1, 2, 3, 4	13*, 14*
3. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual MSIS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
V. RECIRCULATION (RAS)					
A. Sensor/Trip Units					
Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual RAS	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)					
A. Sensor/Trip Units					
1. Steam Generator #1 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG2 > SG1	4	2	3	1, 2, 3	13*, 14*

TABLE [illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)					
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)					
A. Sensor/Trip Units					
1. Steam Generator #2 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG1 > SG2	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VIII. LOSS OF POWER (LOV)					
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

(a) In MODES 3-6, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-6, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two take twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit

1.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2.	Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17- With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia (1)	≥ 1822 psia (1)
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia	≥ 1822 psia
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
Containment Pressure High - High	≤ 8.5 psig	≤ 8.9 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	≥ 919 psia	≥ 912 psia
2. Steam Generator Level - High	≤ ~~≥~~ 91.0% NR(2)	≤ ~~≥~~ 91.5% NR(2)
3. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

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TABLE 3[illegible]ontinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	$\geq$ 8.9% of Span	$\geq$ 8.4% of Span
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	$\geq$ 25.8% WR(3)	$\geq$ 25.3% WR(3)
2. Steam Generator Δ Pressure - SG2 > SG1	$\leq$ 185 psid	$\leq$ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	$\geq$ 25.8% WR(4)	$\geq$ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	$\leq$ 185 psid	$\leq$ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	$\geq$ 3250 volts	$\geq$ 3250 volts
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2930 to 3744 volts with a 35-second maximum time delay	2930 to 3744 volts with a 35-second maximum time dela[y]

TABLE NOTATION 3.3-4

ATTACHMENT 3-23-A

1) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

2) % OF THE DISTANCE BETWEEN STEAM GENERATOR UPPER AND LOWER LEVEL NARROW RANGE INSTRUMENT NOZZLES.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Containment Purge Valve Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. SRAS	
Containment Sump Recirculation	Not Applicable
f. AFAS	
Auxiliary Feedwater Pumps	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure - Low	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.3*/11.7**
c. Containment Isolation	≤ 23.1*/13.1**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.2*/11.7**
c. Containment Isolation	≤ 23.0*/13.0**
d. Main Steam Isolation	≤ 11.0*/11.0**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 31.3*/21.6**
5. Steam Generator Pressure - Low	
a. Main Steam Isolation	≤ 11.1/11.1**
6. Refueling Water Storage Tank - Low	
a. Containment Sump Recirculation	≤ 60.0/60.0**
7. Steam Generator Level - Low	
a. Auxiliary Feedwater (Motor Drive) - SIAS	≤ 26.2*/16.5**
b. Auxiliary Feedwater (Motor Drive) - No SIAS	≤ 26.3*/13.1**
c. Auxiliary Feedwater (turbine drive) - SIAS	≤ 21.1 /21.1**
d. Auxiliary Feedwater (turbine drive) - No SIAS	≤ 21.1 /21.1**
8. Steam Generator Level - High	
a. Main Steam Isolation	≤ 11.0*/11.0**
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator	≤ 23.1*/13.1**

DELETE SEE NEW TABLE

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TABLE 3.3-5 (continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low -	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~96.2~~ 17*/ ~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~51.2~~ 69*/ ~~41.2~~ 59**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~96.2~~ 17*/ ~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~61.2~~ 69*/ ~~41.2~~ 59**
d. Main Steam Isolation	
1. MSIS actuated MSIV's	≤ 6.2*/6.2**
2. MSIS actuated MFIV's	≤ 11.2*/11.2**
3. Other MSIS actuated valves	≤ ~~21.2~~ 69*/ ~~51.2~~ 59**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 33*/23**

AND REVIEW [redacted]

5. Steam Generator Pressure - Low
   a. Main Steam Isolation
      1. MSIS actuated MSIV's ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's ≤ 11.2*/11.2**
      3. Other MSIS actuated valves ≤ ~~21.2~~ 69*/~~21.2~~ 59**

6. Refueling Water Storage Tank - Low
   a. Containment Sump Recirculation ≤ 45*/45**

7. Steam Generator Level - Low
   a. Auxiliary Feedwater (Motor Drive) - SIAS ≤ 45*/30**
   b. Auxiliary Feedwater (Motor Drive) - No SIAS ≤ 45*/30**
   c. Auxiliary Feedwater (Turbine Drive) - SIAS ≤ 45*/30**
   d. Auxiliary Feedwater (Turbine Drive) - No SIAS ≤ 45*/30**

8. Steam Generator Level - High
   a. Main Steam Isolation
      1. MSIS actuated MSIV's ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's ≤ 11.2*/11.2**
      3. Other MSIS actuated valves ≤ ~~21.2~~ 69*/~~21.2~~ 59**

PROOF AND

9. Steam Generator ΔP- High Coincident with Steam Generator Level Low

a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator - $\leq \cancel{31.2\%}$ 45 / $\cancel{21.2\%}$ 30

PROOF AND REVIEW

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	
Loss of Power	≤ 2.4
11. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
Loss of Power 90% system voltage	≤ 35.0
NOTE: Response time for Motor-Driven and Steam-Driven Auxiliary Feedwater Pumps that start on ESF signals on all ESF signal starts	≤ 60

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
I. SAFETY INJECTION (SIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3, 4
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual SIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3) R(1)	1, 2, 3, 4
II. CONTAINMENT ISOLATION (CIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CIAS	NA	NA	M	1, 2, 3, 4
4. Manual SIAS	NA	NA	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
II. CONTAINMENT ISOLATION (Continued)				
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
III. CONTAINMENT SPRAY (CSAS)				
A. Sensor/Trip Units				
1. Containment Pressure -- High - High	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CSAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4

PROOF AND REVIEW

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
IV. MAIN STEAM LINE ISOLATION (MSIS)				
A. Sensor/Trip Unit				
1. Steam Generator Pressure - Low	S	R	M	1, 2, 3, 4
2. Steam Generator Level - High	S	R	M	1, 2, 3, 4
3. Containment Pressure - High	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual MSIS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	R(5) M(1) (2) (3)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
V. RECIRCULATION (RAS)				
A. Sensor/Trip Units				
Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual RAS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	2(1) M(1) (2) (3)	1, 2, 3, 4
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)				
A. Sensor/Trip Units				
1. Steam Gener..or #1 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG2 > SG1	S	R	M	1, 2, 3

TABLE 4.[illegible]ontinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG1 > SG2	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VIII. LOSS OF POWER (LOV)				
A. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	S	R	R	1, 2, 3, 4
B. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	S	R	R	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays ___, ___, ___, ___, ___, ___, and ___ are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

REACTOR COOLANT SYSTEM

PRESSURIZER

AUXILIARY SPRAY

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves to OPERABLE status within [redacted] 72 HOURS [redacted] or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With none of the above required auxiliary spray valves OPERABLE, restore at least one valve to OPERABLE status within the next 6 HOURS [redacted]s or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 The auxiliary spray valves shall be cycled [redacted] at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of ~~2235~~ 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve or within 72 hours following a system response to an Engineered Safety Feature actuation signal.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* TESTING PER SPECIFICATION 4.4.5.2.2.d IS NOT APPLICABLE DUE TO POSITIVE INDICATION OF VALVE POSITION IN THE CONTROL ROOM

1. LEAKAGE RATES LESS THAN OR EQUAL TO 1.0 GPM ARE CONSIDERED ACCEPTABLE

2. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED ACCEPTABLE IF THE LATEST MEASURED RATE HAS NOT EXCEEDED THE RATE DETERMINED BY PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

3. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED UNACCEPTABLE IF THE LATEST MEASURED RATE EXCEEDED THE RATE DETERMINED BY THE PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

4. LEAKAGE RATES GREATER THAN 5.0 GPM ARE CONSIDERED UNACCEPTABLE.

PROOF AND REVIEW

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE	DESCRIPTION
1) SIV 237	LOOP 1A RC/SI CHECK
2) SIV 247	LOOP 1B RC/SI CHECK
3) SIV 217	LOOP 2A RC/SI CHECK
4) SIV 227	LOOP 2B RC/SI CHECK
5) SIV 235	LOOP 1A SIT CHECK
6) SIV 245	LOOP 1B SIT CHECK
7) SIV 215	LOOP 2A SIT CHECK
8) SIV 225	LOOP 2B SIT CHECK
9) SIV 542	LOOP 1A SI HEADER CHECK
10) SIV 543	LOOP 1B SI HEADER CHECK
11) SIV 540	LOOP 2A SI HEADER CHECK
12) SIV 541	LOOP 2B SI HEADER CHECK
13) SIV 522	LOOP 1 HP LONG TERM RECIRCULATION CHECK
14) SIV 523	LOOP 1 HP LONG TERM RECIRCULATION CHECK
15) SIV 532	LOOP 2 HP LONG TERM RECIRCULATION CHECK
16) SIV 533	LOOP 2 HP LONG TERM RECIRCULATION CHECK
17) UV 651 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
18) UV 652 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION
19) UV 653 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
20) UV 654 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 20°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

DATE SUBJECT

DATE

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	38°	1.5	8 - 10
2	43°	1.5	Standby
3	137°	1.5	4 - 5
4	142°	1.5	Standby
5	230°	1.5	12 - 15
6	310°	1.5	18 - 24

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 417 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 255°F during cooldown

b. 295°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 255°F.

b. Heatup with the RCS temperature less than or equal to 295°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

3.4.10 BOTH REACTOR COOLANT SYSTEM VENT PATHS FROM THE REACTOR VESSEL HEAD SHALL BE OPERABLE AND CLOSED.

APPLICABILITY: MODES 1, 2, 3 AND 4

ACTION:

a. WITH ONLY ONE OF THE ABOVE REQUIRED REACTOR VESSEL HEAD VENT PATHS OPERABLE, RESTORE BOTH PATHS TO OPERABLE STATUS WITHIN 72 HOURS OR BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS AND IN HOT SHUTDOWN WITHIN THE FOLLOWING 6 HOURS

b. WITH NONE OF THE ABOVE REQUIRED REACTOR VESSEL HEAD VENT PATHS OPERABLE, RESTORE AT LEAST ONE PATH TO OPERABLE STATUS WITHIN THE NEXT 6 HOURS OR BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS AND IN HOT SHUTDOWN WITHIN THE FOLLOWING 6 HOURS.

SURVEILLANCE REQUIREMENTS

4.4.10 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent valve through at least one complete cycle of full travel from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

PROOF AND REVIEW

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between ~~28%~~ (1802 cubic feet) and ~~72%~~ (1914 cubic feet) level (e.g. BETWEEN 28% AND 72%) as read on narrow range indication), Not accepted

c. A boron concentration between ~~4000~~ 2000 and 4400 ppm of boron, and review

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**.

f. Nitrogen vent valves are capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

†With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 60% wide range indication (1415 cubic feet) and 72% narrow range indication (1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 39% wide range indication (962 cubic feet) and 72% narrow range indication (1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.
2000

c. At least once per 31 days when the RCS pressure is above 700 psig, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 700 psig with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

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3/4.5.2 ECCS SUBSYSTEMS - T_{cold} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1750 psia.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIHV-604	1. HOT LEG INJECTION	1. SHUT
2. SIHV-321	2. HOT LEG INJECTION	2. SHUT
3. SIHV-609	3. HOT LEG INJECTION	3. SHUT
4. SIHV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

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1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

2. Verifying that a minimum total of 464 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

3. Verifying that when a representative sample of 0.055 ± 0.001 lb of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on (SIAS and RAS) test signal(s).

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a. High pressure safety injection pump.

   b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI and LPSI pump minimum bypass recirculation flow line isolation valves close, and the LPSI pumps stop.

f. By verifying that each of the following pumps develops the differential indicated pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1830 psid.

2. Low pressure safety injection pump greater than or equal to 195 psid.

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g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number	Hot Leg Injection Valve Number
1. SI 617, SI 616	1. SI 615, SI 306	1. SI-604
2. SI 627, SI 626	2. SI 625, SI 307	2. SI-609
3. SI 637, SI 636	3. SI 635	3. SI-321
4. SI 647, SI 646	4. SI 645	4. SI-331

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System - Single Pump

1. ~~Injection Leg 1A, equal to 277 ± 5 gpm~~

2. ~~Injection Leg 1B, equal to 277 ± 5 gpm~~

3. ~~Injection Leg 2A, equal to 277 ± 5 gpm~~

4. ~~Injection Leg 2B, equal to 277 ± 5 gpm~~

THE SUM OF THE INJECTION LINE FLOW RATES, EXCLUDING THE HIGHEST FLOWRATE, IS GREATER THAN OR EQUAL TO 816 GPM

LPSI System Single Pump

1. Injection ~~Leg~~ LOOP 1, total flow equal to 4900 ± 100 gpm

2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.

3. Injection ~~Leg~~ LOOP 2, total flow equal to 4900 ± 100 gpm

4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.

Simultaneous Hot Leg and Cold Leg Injection - Single Pump

1. Hot Leg, flow equal to 545 ± 20 gpm

2. Cold Leg, flow equal to 545 ± 20 gpm

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

refueling or shutdown that

CAPS

c. Prior to startup following any cold shutdown of 30 days or longer, by verifying (by means of a flow test) the normal flow path from the condensate storage tank to each of the steam generators through each of the auxiliary feedwater pumps. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump. or MODE 4

d.

delivers 750 gpm at 1250 psig or equivalent

PLANT SYSTEMS

ATMOSPHERE DUMP VALVES

LIMITING CONDITIONS FOR OPERATIONS

3.7.1.6 The atmospheric dump valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*

ACTION:

With less than one atmospheric dump valve per steam generator OPERABLE, restore the required atmospheric dump valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying nitrogen accumulator tank at a pressure $\geq$ 400 PSIG.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer verify that all valves will open and close fully.

* When steam generators are being used for decay heat removal.

3/4.7.13 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, 3, and 4. covered by spec 3.4.1.3

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.13 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot lets, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than ~~370~~ 410 psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than ~~450~~ 500 psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia.

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PROOF AND REVIEW

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

c. Both reactor coolant loops and at least one reactor coolant pump in each loop shall be in operation.

APPLICABILITY: During STARTUP and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, or if less than the above required reactor coolant loops is in operation and circulating reactor coolant, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.3.3 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

3/4.10.8 SAFETY INJECTION TANK PRESSURE

LIMITING CONDITION FOR OPERATION

3.10.8 The safety injection tank (SIT) pressure of Specification 3.5.1d. may be suspended for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b. and pressurized to 175 to 225 psig below the RCS pressure, or NOT TO GO BELOW 254 PSIG

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2 and 3.

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.8, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.8.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.8.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specification 3.5.1b.

b. All the SITs pressures meet the requirements of Specification 3.10.8.

c. The valve alignment from the SITs to the RCS has not changed.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal ~~9.43~~ 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

TABL 1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - ~~T_{avg}~~ Temperature from ≤ ~~200~~ 70°F to ≥ 565°F; cooldown cycle - ~~T_{avg}~~ Temperature from ≥ 565°F to ≤ ~~200~~ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heatup cycle - Pressurizer temperature from < ~~200~~ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ ~~200~~ 70°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psi~~g~~A with RCS temperature between 100°F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one-half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle.
	15000 POWER CHANGE cycles	CYCLES FROM 15% TO 100% FULL LOAD, AT A RATE OF 5% PER MINUTE EITHER INCREASING OR DECREASING (30,000 CYCLES TOTAL)
	10^6 STEP CHANGES OF 100 psi AND 10°F (20°F FOR SURGE LINE)	PRESSURE VARIATIONS BETWEEN THE PRESSURIZER PRESSURE SETPOINT FOR BACKUP HEATER ACTUATION AND SPRAY VALVE OPENING. TEMPERATURE VARIATIONS DUE TO CEA CONTROLLER; 2000 STEP CHANGE OF 10% FULL POWER.
	200 PRIMARY SYSTEM LEAK TEST CYCLES	LEAK TEST PRIMARY SYSTEM AT A PRESSURE OF 2250 PSI AT A TEMPERATURE FROM 120°F to 400°F

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - ~~T_{avg}~~ TEMPERATURE from ≤ ~~200~~ 70°F to ≥ 565°F; cooldown cycle - ~~T_{avg}~~ TEMPERATURE from ≥ 565°F to ≤ ~~200~~ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heatup cycle - Pressurizer temperature from ≤ ~~200~~ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ ~~200~~ 70°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psig with RCS temperature between 120 °F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle.
	15,000 power change cycles	Cycles from 15% to 100% full load, at a rate of 5% per minute, either increasing or decreasing. (30,000 cycles total)
	10^6 step changes of 100 psi and 10°F (20°F for surge line)	Pressure variations between the pressurizer pressure setpoint for backup heater actuation and spray valve opening. Temperature variations due to CEA controller; 2000 step change of 10% full power.
	200 PRIMARY SYSTEM LEAK TEST CYCLES	LEAK TEST PRIMARY SYSTEM AT A PRESSURE OF 2250 PSI AT A TEMPERATURE OF [illegible]

5-~~8~~7A

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TABLE [illegible]-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray Nozzle	UNLIMITED Number of cycles ~~Calculate usage factor per Table 5.7-2.~~	MAIN SPRAY (4 PUMPS OPERATING) Main spray (less than four RCP operating) with fluid $\Delta T_m > 200°F$.
	CALCULATE USAGE FACTOR PER TABLE 5.7-2	Auxiliary spray ~~with fluid $\Delta T_a > 200°F$~~. AT VARIOUS INITIAL FLUID TEMPERATURES WITH FLUID $\Delta T_a \leq 200°F$
		AUXILIARY SPRAY WITH FLUID $\Delta T_a > 200°F$

ΔT_m = THE DIFFERENCE IN TEMPERATURE BETWEEN THE PRESSURIZER AND MAIN SPRAY WATER AS ADJUSTED BY THE INSTRUMENT CORRECTION FACTOR.

ΔT_a = THE DIFFERENCE IN TEMPERATURE BETWEEN THE PRESSURIZER AND AUXILIARY SPRAY WATER AS ADJUSTED BY THE INSTRUMENT CORRECTION FACTOR.

TABLE 5.7-1 (Cont'd)

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Pressurizer Spray Nozzle		Main Spray (4 pumps operating)
	Unlimited number of cycles	Main Spray (Less than 4 pumps operating) - with fluid $\Delta T_m \leq$ ~~X~~°F. 200
		Auxiliary spray at various initial fluid temperatures with fluid $\Delta T_a \leq$ ~~X~~°F. 200
	Calculate usage factor per Table 5.7-2	Main spray (less than 4 pumps operating) with fluid $\Delta T_m >$ ~~X~~°F. 200
		Auxiliary spray with fluid $\Delta T_a >$ ~~X~~°F. 200

ΔT_m = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

ΔT_a = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.

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[illegible] 5.7-2

PRESSURIZER SPRAY NOZZLE USAGE FACTOR

Main Spray				Auxiliary Spray			
ΔT_m	N_A	N	N/N_A	ΔT_a	N_A	N	N/N_A
201-250	7900			201-250	50000		
251-300	4500			251-300	2200		
301-350	2900			301-350	1300		
351-400	1900			351-400	850		
401-450	1200			401-450	550		
451-500	850			451-500	375		
501-550	555			501-550	225		
				551-600	150		
		$\Sigma N/N_A$ =	________			$\Sigma N/N_A$ =	________

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ____________

$\Sigma N/N_A$ (Aux. Spray) ____________

Total ____________ = Cumulative Usage Factor

TABLE [illegible]-2 (Continued)

Where:

$\Delta T_a = (T_{101} - T_{229}) + 60$

$\Delta T_m = (T_{101} - T_{103^* \text{ or } 104^*}) + 70$

NA = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow throughout the pressurizer spray nozzle.

2. If the difference between pressurizer water temperature and the spray water temperature exceeds 200°F each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N by N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

4. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of nozzle fatigue shall be performed within 90 days. The evaluation shall determine that the nozzle remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated.

*Use lower of two temperatures.

3/4.4.3.2 AUXILIARY SPRAY VALVES

Cycling of the auxiliary spray valves during normal plant cooldown will insure the operability of the valves while not causing any additional impact on pressurizer spray nozzle usage factor.

PROOF AND REVIEW

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.231 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 4000 gpm will circulate one equivalent Reactor Coolant System volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to ~~241°F~~ 255°F during cooldown or ~~291~~ 295°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

PROOF AND REVIEW

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SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. BUT BECAUSE OF HIGHER NEUTRON FLUENCE MAY BE CONTROLLING ~~Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location~~.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp. ~~therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined.~~ Consequently, ~~for the cases in which the outer wall of the vessel becomes the stress controlling location,~~ each heatup rate of interest must be analyzed on an individual basis, FOR BOTH THE INNER AND OUTER WALL.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and ~~copper content of the material in question~~ RESIDUAL ELEMENT CONTENT, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 255 ~~241~~°F during cooldown and 295 ~~281~~°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

PROOF AND REVIEW

ECCS SUBSYSTEMS (Continued)

assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

AT ATMOSPHERIC PRESSURE

1. The pressurizer pressure is ~~15 psia~~
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) ~~3.2~~ 6.4 gpm (to/from) the ~~2450~~ 4900 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

DELETE REPLACE WITH ATTACHED PAGE

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1381 psig) of its design pressure of 1256 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 19.53×10^6 lb/hr which is 113% of the total secondary steam flow of 17.18×10^6 lb/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two-loop, or four-pump operation

$$SP = \left(\frac{10-N}{10}\right) \times 113$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = total number of secondary safety valves for one steam generator.

N = the number of inoperable secondary safety valves on the steam generator with the greater number of inoperable valves.

113 = the ratio of the total relieving capacity of all twenty (20) secondary safety valves (19.53×10^6 lb/hr at 1355 psig, maximum set pressure plus 3%, accumulation) over the secondary steam flow at 100% Rated Thermal Load (17,180,000 lb/hr).

The main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes, a turbine trip (without reactor trip or cutback) from RATED THERMAL POWER with a coincident loss of condenser heat sink (i.e., no steam bypass) is assumed. The combined relieving capacity of the pressurizer safety valves, and the heat removal capacity of the MSSVs, is sufficient to maintain the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks and 110% of design pressure for all other events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity of all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity of 19.53×10^6 lbm/hr given in Table 3.7-1 as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainity) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity of 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

Although the variable high power reactor trip is not relied on for the limiting overpressure events, the ceiling on this trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

SP = Allowable Power Level + 9.8

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is the ratio of the available relieving capacity of the total steam flow at rated power

10 = total number of main steam safety valves for one steam generator

N = number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves

109.2= ratio of main steam safety valve relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainity (see above text)

9.8= BAND between the maximum thermal power and the variable overpower trip setpoint ceiling

PROOF AND REVIEW

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ~~with the~~ ENSURES THAT A minimum WATER volume OF 300,000 GALLONS. ~~ensures that sufficient water~~ is available to maintain the Reactor Coolant System at HOT STANDBY for ~~4~~ 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the nitrogen accumulator at a pressure ≥ 400 psig is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV which holds the plant at hot standby while dissipating core decay heat or which allows a flow of sufficient steam to maintain a controlled reactor cooldown rate. A pressure of 400 psig retains sufficient nitrogen volume for 4 hours of operation at hot standby plus 6.5 hours of operation to reach cold shutdown under natural circulation conditions in the event of failure of the normal control air system.

ICSB

ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix, Arizona 85036

received 9/19/84

JND

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

8411020371

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

PDR ~~8410020130~~

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt /TBK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc

cc: E. Licitra (w/a)
J. B. Martin (w/a)
R. Zimmerman (w/a)
G. Fiorelli (w/a)

PROOF AND REVIEW

PG 1-9 Add # note to startup Not accepted

JUSTIFICATION:

The 3.10.5 is for the first core only but footnote on 1-9 remains forever

PVNGS Operations and engineering personnel needs this change to clarify and locate the special test exception for initial criticality and low power physics testing. Since this condition is not "Normal" Operating Procedure when the operations go critical below 350°F they may try to take some kind of corrective action to comply with Tech Specs not realizing that they are operating in a special test exception area. This change will assist Resident Inspectors identify and locate a Special Test Exception Factor.

PG 2-3 TABLE 2.2-1 ITEM 1.A.7.c ICSB RSB

Change allowable value to 42.1%.

JUSTIFICATION:

New CE number.

PG 2-5 ADD to FOOTNOTE 6 ICSB RSB

Setpoints are % of 100% power flow conditions.

JUSTIFICATION:

CE words to clarify and explain what exactly is happening.

PG B2-4 ADD TO STEAM GENERATOR LEVEL - LOW

...10 minutes before auxiliary feedwater is required to prevent degraded core cooling RSB

JUSTIFICATION:

PVNGS uses the term auxiliary vs emergency emergency as shown in the Tech Spec. Also the added phrase provides more detailed description as to what is really happening.

PG B2-5 CHANGE DNBR-LOW

...Floor of 1861 psia. ~~RSB~~ CPB

JUSTIFICATION:

CE changed number

PROOF AND REVIEW ICSB

RSB

CPS

PG B2-6 **CHANGE:**

See page.

JUSTIFICATION:

CE changed numbers.
New descriptions and additions for Steam Generator Level High and Reactor Coolant Flow - Low provide more accurate and detailed information to assist the Operating Personnel.

ICSB

CPB

RSB

PG B2-7 **CHANGE:**

See page.

JUSTIFICATION:

Again, to provide a more accurate statement for CPC Addressable Constants.

PG 3/4 1-1 **CHANGE ACTION STATEMENT**

See page.

~~RSB~~ CPB

JUSTIFICATION:

This change has been continuously discussed with the NRC. As the spec is presently written there is alot of operator confusion as to if the operator immediately initiates boration or goes to Spec. 3.1.3.6 which allows the operator to have two hours to take corrective action. Many of our Operations People have some confusion which will lead them to taking various corrective actions. The proof and review LCO 3.1.1.1 is in conflict with LCO 3.1.3.6. The action statement in LCO 3.1.1.1 requires immediate boration to reestablish 6% delta K/K shutdown margin, if violated. However, LCO 3.1.3.6 allows two (2) hours to recover rod position above the Transient Insertion Limit of Figure 3.1.3 or Figure 3.1.4. Since Transient Insertion Limits on these curves represents a value of available shutdown margin equivalent to 6% delta K/K, this means there are two different action requirements for exceeding the same limit. In actuality, the only verification of shutdown margin that is required to be performed during normal critical operation is the CEA position verification.

PG 3/4 2-5 CHANGE LCO and Surveillance - 4.2.4.4 LPB

JUSTIFICATION: Table 3.3-1 to be submitted later

Action 6 applies to this LCO not just 6C as stated. Also change the LCO to say ...Table 3.3-1. The rest of the sentence can be deleted because it is included in Action 6.

Change 4.2.4.4 to ...once per 31 EFPD because this surveillance refers to a burnup period vs. a calandar period. This change is in compliance with the way PVNGS intends to operate. If the unit is shutdown for any period of time and nothing was done to change the COLSS or CPC DNBR parameters/calculations they do not need to be verified.

PGS 3/4 2-~~9~~ 7 THESE GRAPHS WILL BE SUPPLIED IN ABOUT A WEEK LPB
3/4 2-10

PGS 3/4 3-7, CHANGE:
3/4 3-8,
3/4 3-7a, See attached. RSB ICSB
3/4 3-8b

JUSTIFICATION:

Current Arizona Unit 1, Cycle 1 Technical Specifications does not specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEAC's out of service. An analysis was done to specify this lower power level.

This work is the completion of the CEAC's OOS work. This analysis improves ANPP Unit 1, Cycle 1 power capability from about 75% to greater than about 90% with both CEAC's out of service. The analysis of the documents and quality assures this result.

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEAC's out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature	:	580°F
Low Pressure	:	1785 psia
ASI	:	-.3
Underflow fraction	:	0.865
Low Flow	:	95% of full flow
High Radial Peak	:	1.70 (Bank 5+4+PLR; PDIL = 40% Power; Reference 3)

Conclusion:

This revised Technical Specifications include a minimum power (50% of rated power) below which an additional power reduction is unnecessary, if there is a CEA misalignment with CEAC's out of service.

The minimum power for POL is 70%.

The added statement to action 6a is justified to the justification section for 3/4 1-17.

The work disabled in Items 6.b.1.c and 6.C.1.b are to be replaced with "Placed out of Service". This is consistent with PVNGS Terminology and also there is an indicator on the control panel that states "Out of Service" for the reactor power cutback system.

Delete the terms in 6.b.2.b and 6.C.2.b "...The inoperable status" to read ..."Be indicated that both CEACs are inoperable". This will avoid operator confusion and is consistent with how PVNGS Operators refer to this situation.

PG 3/4 3-9 CHANGE: ICSB RSB

See Page.

JUSTIFICATION:

CE number changes.

PG 3/4 3-10 DELETE:

Current # note add new one. See page. RSB

JUSTIFICATION: ICSB

The new pound note had been proposed to the NRC for the past 2 years. This change is consistent and reflects the way PVNGS will test the response times. This change is also agreed upon by CE.

PROOF AND REVIEW

"The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input."

The Proof and Review copy, # note reads, "Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown." This requires pertubation of the reactor plant and excludes the capability to utilize test equipment to verify response time. Pertubating the plant is an unnecessary safety and radiological (crud) risk.

PG 3/4 3-10a Type, see page. CPB RSB ICSB

PG 3/4 3-12 CHANGE:

III B to read change functional test R. RSB ICSB

JUSTIFICATION:

Page 3/4 3-12, Table 4.3-1 III. B., Channel Functional Test column. Change to read: "R" (delete "M, S/U(1)").

This change is consistent with the SONGS 2 and SONGS 3 Tech Specs.

Table 4.3-1 III. A. provides adequate surveillance requirements for the Reactor Trip Breakers. The difference in plant equipment hardware between Table 4.3-1III. A. and III. B. are the reactor trip pushbutton switches. A surveillance test frequency of "R" is adequate for operability verification of pushbutton switches.

PG 3/4 3-13 ADD

Statement, see page. RSB ICSB

JUSTIFICATION:

Add the statement to note 8 for clarification and providing an alternate and more detailed method as to how to determine RCS flow rate.

PG 3/4 3-22 ADD
3-23,
Attachment See page.
3-23-A

JUSTIFICATION: RSB ICSB

Add the footnotes as indicated to provide detailed information to the plant staff as what actions to take to avoid confusion. The NR and WR identify if the readings are taken from the Wide Range or Narrow Range instrument.

accepted as is me

PG 3/4 3-24 Typo.

PG 3/4 3-25 DELETE:

CSB
RSB
ICSB

Old page. Insert new table.

JUSTIFICATION:

The new table is a more useable table to the operators. It breaks the systems down to where everyone understands. The old table was confusing in that the times represented showed the TOTAL time of the System response times not done of the key subsystem response times such as shown in the new table.

PG 3/4 3-26 DELETE:

RSB
ICSB

Note. See page.

JUSTIFICATION:

The response times for the AFWP, are shown in the body of the table. Showing a note with another response time doesn't match is not really necessary. We have had alot of questions generated as to why the note if the AFWP, response times are in the Body of the table. Operations are continuously confused by both values; they want only 1 to worry about.

PG 3/4 3-27, 28, 29, 30, 31, 32 Change automatic actuation logic channel functional test to R

CSB
RSB
ICSB

JUSTIFICATION:

PVNGS will be submitted a letter to the NRC next week providing detailed justification. In summary we do not want to test this logic at power operation because various equipment would actuate and could trip the unit. Many other utilities are having problems meeting this Tech. Spec. and are going to be asking for a Tech. Spec change, in fact one set in there proposed change and justification about a month ago.

PG 3/4 3-34, 35 CHANGE:

~~NMEB~~
METB

Table 3.3-6. See page.

JUSTIFICATION:

Incorporation of RETS ("Radiological Effluent Technical Specifications") into the STS ("Standard Technical Specifications") has resulted in duplication of operability requirements for the Effluent Monitoring System.

PROPOSED

PG 3/4 4-32 CHANGE:

LCO and Action Item b. See page.

RSB

JUSTIFICATION:

CE Number Change.

PG 3/4 5-1
5-2
CHANGE:

LCO b. See Page.

RSB
ICSB

JUSTIFICATION:

To clarify the Spec. for operations to avoid further confusion. Also the 2000 ppm change is based on a new CE Number.

PG 3/4 5-5 CHANGE:

(a). See page.

RSB

JUSTIFICATION:

CE new number.

PG 3/4 5-6 CHANGE:

HPSI System - Single Pump. See page.

RSB

JUSTIFICATION:

This specification needs to be changed to comply with how we tested the pump during startup as well as how we will test the HPSI pump during operations. This change is in compliance with CE's design criteria.

Change LPSI Pump Sections 1 and 3 from leg to loop.

PG 3/4 6-1 CHANGE:

Surveillance 4.6.1.1a. Table 3.6-0. See pages.

CSB

JUSTIFICATION:

Operations Department requests the attached changes to the PVNGS Technical Specification. Justification for each proposed change is summarized as follows:

Technical Specification 3/4.6.1.1. "Containment Integrity" and Technical Specification 3/4.6.3, "Containment Isolation Valves."

PROOF AND REVIEW

PG 3/4 6-12, 13 and 14 — **DELETE:** SGEB

See page.

JUSTIFICATION:

These pages applied to the Old Tondon Tech. Spec. not to the Tech. Spec. in the proof/review copy. It should have been deleted last revision. The deleted information in the old tables shows up in our new table.

PG 3/4 6-15 — **CHANGE:** CSB

Action C. See page.

JUSTIFICATION:

Add the following ... OPERABLE status "or isolate the penetrations" within 24 hours ...

This addition is to bring Action C into compliance with Actions A and B.

PG 3/4 6-16 — Typo's. Surveillance Req. 4.6.2.1.C.

PG 3/4 6-20, 6-21 — **CHANGE:** LCO - Add * Asterisk. CSB

LCO - Add * Asterisk. See page.

JUSTIFICATION:

The statement was added to clarify to the operators exactly what needs to be done. This change was prompted by a difference in interpretation between our region and an operating utility.

PG 3/4 6-22, 6-35 — **CHANGE:** CSB

Table 3.6-1. See page.

JUSTIFICATION:

See Justification provided in 3.6-1.

PG 3/4 6-36 — **CHANGE:**

Action Statements. See page.

CSB

JUSTIFICATION:

The action of this Tech. Spec. based on problems at SONGS. These proposed and approved Tech. Spec. provides

PROOF AND REVIEW

- a more detailed action statement to instruct the operators as to what they need to do in situations not identified in the old action statement.

PG 3/4 6-37 CHANGE:

Surveillance 4.6.4.2.b.1. Delete all.

me

JUSTIFICATION:

The word "ALL" needs to be deleted. There is some instrumentation that is not "vital" or needed to insure proper operation of the recombiner to perform its intended safety functions.

PG 3/4 6-38 CHANGE:

ICSB METB

LCO Delete the last six words of the LCO ... "In each of the three units."

JUSTIFICATION:

This statement is confusing in its original form. Our people interpret it to mean that power needs to be supplied from all three units no matter where the containment hydrogen purge cleanup system is located. The proposed change clarifies this problem.

PG 3/4 6-39 ADD:

METB

Page. This page was deleted in the Proof/Revision copy of the Tech. Spec.

PG 3/4 7-2,3 CHANGE:

ASB

See page.

JUSTIFICATION

CE supplied input.

PG 3/4 7-4 CHANGE:
7-5

ICSB
ASB

Surveillance Requirements 4.7.1.2.a.1, 4.7.1.2.a.3, and 4.7.1.2.c. See pages.

JUSTIFICATION:

Delete the last sentence in 4.7.1.2.1.1. This statement will be broken out into item 4.7.1.2.d. This is done to add clarification.

PG 3/4 7-1

me

PG 3/4 7-39 CHANGE: CEB

Surveillance 4.7.11.6.b. See page.

JUSTIFICATION:

Delete ... "and verifying that the hydrant barrel is dry "... This Spec is used for those plants in climates where freezing occurs. PVNGS, as discussed in the FSAR, is not subject to climates or weather that would cause water in the hydrant barrel from freezing and causing various damage.

PG 3/4 7-41 CHANGE: CEB

LCO and Surveillance 4.7.12.1.b. See page.

JUSTIFICATION:

Delete the fire windows reference. PVNGS doesn't have fire windows.

PG 3/4 7-42 CHANGE: CEB

Surveillance 4.7.12.2. See page.

JUSTIFICATION:

Deletion of Item A is justified in that PVNGS doesn't have any fire door supervision system as described; therefore, we cannot comply with this.

PG 3/4 7-43 CHANGE: ICSB PSB

Surveillance 4.7.13.b. See page.

JUSTIFICATION:

CE new numbers.

PG 3/4 8-1 CHANGE:
8-2
8-7

Tech. Spec. See pages. PSB ORAB

JUSTIFICATION:

The Tech. Spec. was changed to comply with NRC Generic letter 84-15.

Technical Specification 3/4.8.1, "A.C. Sources"

The proposed changes to Tech. Spec. 3/4.8.1 are a result of the applicable recommendations of NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability" dated July 2, 1984.

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1. The trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. AFTER 7 DAYS, OPERATION MAY CONTIN PROVIDED THAT THE CONDITIONS OF ACTION ITEM 6.b OR 6.c ARE MET

b. With both CEACs inoperable and COLSS in ~~operation~~, SERVICE operation may continue provided that:

1. Within 1 hour:

~~a)~~ The margins required by Specification 3.2.4 are increased and maintained at a value equivalent to or greater than the percentage of RATED THERMAL POWER shown on Figure 3.3-1.

c) The Reactor Power Cutback System is ~~disabled~~. PLACED OUT OF SERVIC

A) OPERATION IS RESTRICTED TO THE LIMITS SHOWN IN Fig 3.3-1. THE DNBR MARGIN REQUIRE BY SPECIFICATION 3.2.4 IS REPLACED BY THIS RESTRICTION WHEN BOTH CEAC'S ARE INOPERABLE AND COLSS IS IN OPERATION

b) THE LINEAR HEAT RATE MARGIN REQUIRED BY SPECIFICATION 3.2.1 IS MAINTAINED

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status.~~ BE INDICATE THAT BOTH CEACS ARE INOPERABLE

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

A. FOLLOWING A CEA MISALIGNMENT, WITH BOTH CEAC'S INOPERABLE AND COLSS IN OPERATION, OPERATION MAY CONTINUE PROVIDED THAT:
WITHIN 1 HOUR;
a) THE POWER IS REDUCED TO 81% OF THE PRE-MISALIGNMENT POWER BUT NEED NOT BE REDUCED TO LESS THAN 60% OF RATED POWER.
b) REFER TO SECTION 3.1.3, MOVABLE CONTROL ASSEMBLIES, FOR FURTHER SPECIFICATIONS ON CEA MISALIGNMENT.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

1. Within 1 hour:

   a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by ~~1.18~~ 1.19 and the resulting value is re-entered into the CPCs.

   b) The Reactor Power Cutback System is ~~disabled~~ PLACED OUT OF SERVICE.

   c) THE COLSS OUT OF SERVICE LIMIT LINE, SECTION 3.2.- IS NOT APPLICABLE TO THIS MODE OF OPERATION.

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status~~. BE INDICATED THAT BOTH CEAC'S ARE INOPERABLE

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment, with both CEACs and COLSS inoperable, operation may continue provided that:
within 1 hour:
a) The power is reduced to 80% of the 'pre-misaligned' power, but need not be reduced to less than 50% of rated power.
b) Refer to Section 3.1.3, Movable Control Assemblies, for further Specifications on CEA misalignment

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - Wi : the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 1.15 seconds
2. Pressurizer Pressure - Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ ~~0.75~~ second 0.65
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. Cold Leg Temperature	≤ ~~0.81~~ second## 0.75
d. Hot Leg Temperature	≤ ~~0.81~~ second## 0.75
e. Primary Coolant Pump Shaft Speed	≤ ~~0.52~~ second# 0.75
f. Reactor Coolant Pressure from Pressurizer	≤ ~~0.48~~ second### 0.75
g. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 1.15 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

PROOF AND REVIEW

RECEIVED AUG 1984 NUCLEAR OPERATIONS LICENSING

TABLE [illegible] (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

[#Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown.]

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 13 seconds. Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accomodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

THE PULSE TRANSMITTERS MEASURING PUMP SPEED ARE EXEMPT FROM RESPONSE TIME. TESTING. THE RESPONSE TIME SHALL BE MEASURED FROM THE PULSE SHAPE INPUT

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	D (2, 4), R (4, 5)	M, R (6)	1, 2
9. DNBR - Low	S	D (2, 4), R (4, 5) M (8), S (7)	M, R (6)	1, 2
B. Excore Neutron Flux				
1. Variable Overpower Trip	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5 and *
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M, S/U (1) R	1, 2, 3*, 4*, 5*

PROOF AND REVIEW

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust he CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

EITHER USING THE REACTOR COOLANT PUMP DIFFERENTIAL PRESSURE INSTRUMENTATION AND THE ULTRASONIC FLOW METER ADJUSTED PUMP CURVES (CONSERVATIVELY COMPENSATED FOR MEASUREMENT UNCERTAINTIES) OR

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia (1)	≥ 1822 psia (1)
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia	≥ 1822 psia
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
Containment Pressure High - High	≤ 8.5 psig	≤ 8.9 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	≥ 919 psia	≥ 912 psia
2. Steam Generator Level - High	≤ ~~≥~~ 91.0% NR(2)	≤ ~~≥~~ 91.5% NR(2)
3. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

TABLE 3.[illegible]ontinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	≥ 8.9% of Span	≥ 8.4% of Span
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	≥ 25.8% WR(3)	≥ 25.3% WR(3)
2. Steam Generator Δ Pressure - SG2 > SG1	≤ 185 psid	≤ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	≥ 25.8% WR(4)	≥ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	≤ 185 psid	≤ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3250 volts	≥ 3250 volts
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2930 to 3744 volts with a 35-second maximum time delay	2930 to 3744 volts with a 35-second maximum time dela[y]

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TABLE NOTATION 3.3-4

ATTACHMENT 3-23-A

1) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

2) % OF THE DISTANCE BETWEEN STEAM GENERATOR UPPER AND LOWER LEVEL NARROW RANGE INSTRUMENT NOZZLES.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure - Low	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.3*/11.7**
c. Containment Isolation	≤ 23.1*/13.1**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.2*/11.7**
c. Containment Isolation	≤ 23.0*/13.0**
d. Main Steam Isolation	≤ 11.0*/11.0**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 31.3*/21.6**
5. Steam Generator Pressure - Low	
a. Main Steam Isolation	≤ 11.1/11.1**
6. Refueling Water Storage Tank - Low	
a. Containment Sump Recirculation	≤ 60.0/60.0**
7. Steam Generator Level - Low	
a. Auxiliary Feedwater (Motor Drive) - SIAS	≤ 26.2*/16.5**
b. Auxiliary Feedwater (Motor Drive) - No SIAS	≤ 26.3*/13.1**
c. Auxiliary Feedwater (turbine drive) - SIAS	≤ 21.1 /21.1**
d. Auxiliary Feedwater (turbine drive) - No SIAS	≤ 21.1 /21.1**
8. Steam Generator Level - High	
a. Main Steam Isolation	≤ 11.0*/11.0**
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator	≤ 23.1*/13.1**

DELETE SEE NEW TABLE

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TABLE 3.3-5 (continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low -	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~96.2~~ 17* / ~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~51.2~~ 69* / ~~41.2~~ 59**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~96.2~~ 17* / ~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~51.2~~ 69* / ~~41.2~~ 59**
d. Main Steam Isolation	
1. MSIS actuated MSIV's	≤ 6.2*/6.2**
2. MSIS actuated MFIV's	≤ 11.2*/11.2**
3. Other MSIS actuated valves	≤ ~~21.2~~ 69* / ~~21.2~~ 59**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 33*/23**

PROOF AND REVIEW

5. Steam Generator Pressure - Low
   a. Main Steam Isolation
      1. MSIS actuated MSIV's ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's ≤ 11.2*/11.2**
      3. Other MSIS actuated valves ≤ ~~21.2~~ 69*/~~21.2~~ 59**

6. Refueling Water Storage Tank - Low
   a. Containment Sump Recirculation ≤ 45*/45**

7. Steam Generator Level - Low
   a. Auxiliary Feedwater (Motor Drive) - SIAS ≤ 45*/30**
   b. Auxiliary Feedwater (Motor Drive) - No SIAS ≤ 45*/30**
   c. Auxiliary Feedwater (Turbine Drive) - SIAS ≤ 45*/30**
   d. Auxiliary Feedwater (Turbine Drive) - No SIAS ≤ 45*/30**

8. Steam Generator Level - High
   a. Main Steam Isolation
      1. MSIS actuated MSIV's ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's ≤ 11.2*/11.2**
      3. Other MSIS actuated valves ≤ ~~21.2~~ 69*/~~21.2~~ 59**

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9. Steam Generator ΔP - High Coincident with Steam Generator Level Low

a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator. $\leq$ ~~31.2*~~ 45 / ~~21.2~~ 30

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	
Loss of Power	≤ 2.4
11. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
Loss of Power 90% system voltage	≤ 35.0

NOTE: Response time for Motor-Driven and Steam-Driven Auxiliary Feedwater Pumps that start on ESF signals on all ESF signal starts ≤ 60

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
I. SAFETY INJECTION (SIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3, 4
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual SIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3) 2	1, 2, 3, 4
II. CONTAINMENT ISOLATION (CIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CIAS	NA	NA	M	1, 2, 3, 4
4. Manual SIAS	NA	NA	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
II. CONTAINMENT ISOLATION (Continued)				
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
III. CONTAINMENT SPRAY (CSAS)				
A. Sensor/Trip Units				
1. Containment Pressure -- High - High	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CSAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
IV. MAIN STEAM LINE ISOLATION (MSIS)				
A. Sensor/Trip Unit				
1. Steam Generator Pressure - Low	S	R	M	1, 2, 3, 4
2. Steam Generator Level - High	S	R	M	1, 2, 3, 4
3. Containment Pressure - High	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual MSIS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	R(4) M(1) (2) (3)	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
V. RECIRCULATION (RAS)				
A. Sensor/Trip Units				
Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual RAS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)				
A. Sensor/Trip Units				
1. Steam Generator #1 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG2 > SG1	S	R	M	1, 2, 3

TABLE 4.[illegible] ontinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG1 > SG2	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VIII. LOSS OF POWER (LOV)				
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	S	R	R	1, 2, 3, 4
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	S	R	R	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

1 ~~(2)~~ Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays ___, ___, ___, ___, ___, ___, and ___ are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

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3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between ~~28%~~ (1802 cubic feet) and ~~72%~~ (1914 cubic feet) level as read on narrow range indication (E.G. BETWEEN 28% AND 72%),

c. A boron concentration between ~~4000~~ 2000 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**

f. Nitrogen vent valves are capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

†With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 60% wide range indication (1415 cubic feet) and 72% narrow range indication (1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 39% wide range indication (962 cubic feet) and 72% narrow range indication (1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.

c. At least once per 31 days when the RCS pressure is above 700 psig, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 700 psig with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a) With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

b. WITH 2 HYDROGEN MONITORS INOPERABLE, RESTORE AT LEAST ONE MONITOR WITHIN 48 HOURS OR BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS

c. WITH ONE HYDROGEN MONITOR INOPERABLE, THE PROVISIONS OF SPECIFICATION OF 3.0.4 ARE NOT APPLICABLE

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LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each AFW pump, and the manual discharge line valve of each AFW pump are locked, ~~in the open position~~ SEALED, OR OTHERWISE SECURED IN THE OPEN POSITION.

*Until the steam generators are no longer required for heat removal.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

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c. Prior to startup following any cold shutdown of 30 days or longer, by verifying (by means of a flow test) the normal flow path from the condensate storage tank to each of the steam generators through each of the auxiliary feedwater pumps. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump. or MODE 4

d.

3/4.7.13 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.13 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot lets, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than ~~370~~ psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than ~~450~~ psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia. 410 500

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

INFORMAL

ENCLOSURE 1

ICSB COMMENTS AND RECOMMENDED CHANGES FOR THE PALO VERDE - UNIT 1 TECHNICAL SPECIFICATIONS

1. Technical Specification Section 2.2 (Subsection 2.2.2)

The Palo Verde Technical Specifications contain limiting safety system settings for core protection calculator (CPC) addressable constants as discussed in the Palo Verde SER (NUREG-0857) Section 7.2.5 (item 2). Technical Specification Table 2.2-2 identifies only Type I values. It is the staff's understanding that CE plants utilizing CPCs typically have both Type I and Type II addressable constants. Therefore, the staff recommends that Type II addressable constants be included in the Palo Verde Technical Specifications or justification provided for their omission. Please note that the Core Performance Branch (CPB) will verify the addressable constant allowable values proposed in the Technical Specifications. This has been discussed with CPB.

2. Technical Specification 2.2 (Table 2.2-1)

The Palo Verde FSAR Section 7.2.2.1 refers to CESSAR Section 7.2.2.1. The subject CESSAR section references Table 7.2-4 which gives the applicable nominal trip setpoints for the RPS. A comparison of CESSAR Table 7.2-4 values with those of Table 2.2-1 of the Palo Verde Technical Specifications revealed a number of discrepancies. Also, the values shown in Palo Verde FSAR Section 7.2.2.1 (modification to CESSAR Table 7.2-4) are in disagreement with the Technical Specification setpoints. The applicant should provide sufficient information to resolve these discrepancies.

3. <u>Technical Specification 2.2 (Table 2.2-1)</u>

The Palo Verde FSAR Section 7.2.2.3.2 refers to CESSAR Section 7.2.2.3.2 for equipment design criteria. The subject CESSAR section references Table 7.2-1 for identification of RPS bypasses. Note (5) of the table notations for Technical Specification Table 2.2-1 states that the high local power density and low DNBR trips may be manually bypassed below 1% of Rated Thermal Power whereas CESSAR Table 7.2-1 states that a bypass for these trips can be implemented only if power is below 10^{-4}%. The applicant should provide information to clarify this apparent discrepancy. Please note that this applies to note (c) of Table 3.3-1 also.

4. <u>Technical Specification 2.2 (Basis 2.2.1)</u>

The staff recommends that the Bases Section 2.2.1 on Reactor Trip Setpoints be expanded to include a discussion on the setpoint methodology program used to develop the various protection system trips. The additional information should describe what the setpoint calculations are based upon, etc.

5. <u>Technical Specification 2.2 (Basis 2.2.1)</u>

The Technical Specification Basis states that the low DNBR trip incorporates a low pressurizer pressure floor of 1785 psia. This is

inconsistent with CESSAR Table 7.2-4 which shows the low floor pressure to be 1750 psia. The applicant should provide information to clarify this apparent discrepancy since the PVNGS FSAR references the CESSAR docket for this trip setpoint value.

6. Technical Specification 2.2 (Basis 2.2.1)

The discussion on low DNBR references the "calculated" core DNBR whereas the CE Standard Technical Specifications (STS) and other various CE plant technical specifications reference the "actual" core DNBR. Please provide information to differentiate between the two references and to explain the deviation from the CE-STS as noted above.

7. Technical Specification 2.2 (Basis 2.2.1)

The discussion on low reactor coolant flow states that this trip provides protection against the two pump opposite loop flow coastdown event. CESSAR (referenced by the Palo Verde FSAR) Section 7.2 discusses this trip function as being applicable to only a reactor coolant pump sheared shaft event. The applicant should provide information to clarify this discrepancy.

8. Technical Specification 3/4.1.3 (Subsection 4.1.3.1.2)

Specification 4.1.3.1.2 should be revised to agree with Specifications 3.1.3.2 and 4.1.3.2 (i.e., should reference 5.2 inches).

9. Technical Specification 3/4.1.3 (Subsection 3.1.3.3)

The sentence notated with an (*) should be revised as follows: "Only if the reactor trip breakers happen to be in the closed position."

This modification is not a change but a clarification.

10. Technical Specification 3/4.2.6 (Subsection 3.2.6)

Technical Specification 3.2.6 refers to Figure 3.2-3 which shows the area of acceptable operation for reactor coolant loop cold leg temperature (Tc). Limiting values for Tc are also provided in Basis 2.2.1 as part of the low DNBR discussion. These Basis values appear to be outside of the range of values given in Specification 3.2.6. The ICSB recommends that the applicant provide information to clarify this apparent discrepancy. The information should discuss the correlation between the Technical Specification sections referenced above.

11. Technical Specification 3/4.3.1 (Subsection 4.3.1.4)

Palo Verde references CESSAR Section 7.2.1.1 for system description information. The subject CESSAR section states that optical isolation is utilized at each CEAC output to the CRT display generator. Failures of the optical isolators could seriously compromise the ability of the CEACs to perform the required protective function. The current surveillance requirements do not include provisions

for verifying that the isolation characteristics of these devices has not failed. This is a deviation from the CE-STS. Also, the staff has required all CE plants utilizing core protection calculators system designs to include such technical specification requirements. Therefore, it is the staff's position that periodic tests to verify the isolation characteristics of those isolation devices used to ensure channel independence should be performed.

However, if it can be shown that the optical isolators utilized on Palo Verde are identical to those incorporated on other CE plants and that these optical isolators have been shown to be reliable based on operational history through periodic testing, then the staff will consider withdrawing such a surveillance requirement. Sufficient information would have to be provided to support such a withdrawal.

Also, CESSAR SER (NUREG-0852) Section 7.2.1.3 states that fiber optics with a minimum distance of three feet are used in place of optical isolators for the interface between the two CEACs and the four CPCs. Such fiber optic links do not require periodic testing as do the optical isolators. The staff requests that the applicant verify that the Palo Verde design utilizes (for the CEAC/CPC interface) the fiber optic links as described in the CESSAR SER. If optical isolators are used, then appropriate surveillance requirements should be

incorporated into the Palo Verde Technical Specifications as required by the CE-STS or justification should be provided.

12. Technical Specification 3/4.3.1 (Subsection 4.3.1.5)

Review of Specification 4.3.1.5 revealed that exemptions are to be taken to the totaling of restarts associated with each core protection calculator (CPC) within a 12 hour period. This totaling determines the operability of the CPCs. It is not typical, as noted in the CE-STS and with other CE STS plant reviews, to take such exemptions. The applicant should, therefore, submit information to justify this deviation. Note that this is also related to Action 7 of Table 3.3-1 of the Technical Specifications.

13. Technical Specification 3/4.3.1 (Tables 3.3-1 and 4.3-1)

Notation (*) for Tables 3.3-1 and 4.3-1 should be modified for clarification as follows:

"Only if the protective system trip breakers happen to be in the closed position, the CEA drive system is capable of CEA withdrawal, and fuel is in the reactor vessel."

14. Technical Specification 3/4.3.1 (Table 3.3-1)

Functional Unit III.B. (manual reactor trip) of Table 3.3-1 references Action statement 5. Typically, Action statement 1 is applied

as shown in the CE-STS as well as other CE STS plants. Since Action 5 appears to be less conservative than Action 1 and is a deviation from the CE-STS, the staff's position is that Action 1 should be referenced for the manual trip function unless sufficient information is provided to justify this deviation.

15. Technical Specification 3/4.3.1 (Table 3.3-1)

Item 3 (steam generator pressure-low) of Actions 2 and 3 refers to steam generator level 1-low and steam generator level 2-low as functional units to be bypassed or tripped. The applicant should provide information to correlate these functional units with the differential pressure (ΔP) functional units incorporated to determine a faulted or intact steam generator for control of emergency feedwater actuation. The staff recommends that a footnote be added to Table 3.3-1 to clarify this correlation. Please note that this also applies to Actions 13 and 14 of Table 3.3-3, items 1. and b.1., respectively.

16. Technical Specification 3/4.3.1 (Table 3.3-1)

Action 6 of Table 3.3-1 provides operational guidance when the CEAC(s) becomes inoperable. Typically, the operator is required to initiate either a trip or bypass function when a channel becomes inoperable. However, this is not the case for Action 6. Instead, the statement allows plant operation to continue provided that various actions are

taken related to limiting conditions for operation addressed in Technical Specification Sections 3/4.1.3 and 3/4.2.4. These specifications are not within ICSB's scope of review. Therefore, ICSB recommends that the appropriate branch(es) be contacted by SSPB to ensure that the review of Specifications 3/4.1.3 and 3/4.2.4 includes Action 6 of Table 3.3-1.

It should be noted that item b.1 of Action 6 deviates from the CE-STS in that it does not reference Specification 3.2.1 which falls in the same category as Specifications 3/4.1.3 and 3/4.2.4 addressed above. Thus, the appropriate review branch should consider this during their review of this issue. Also, the responsible review branch should be made aware that item b. of the Action statement for Specification 3.2.1 is less conservative than that of the CE-STS.

17. <u>Technical Specification 3/4.3.1 (Table 3.3-2)</u>

Notation (**) to Table 3.3-2 provides operational guidance for response times associated with CEA positions. This note references Specification 3.1.3.4 which is not within ICSB's scope of review. Therefore, ICSB recommends that the appropriate branch be contacted by SSPB to ensure that the review of Specification 3.1.3.4 is correlated with note (**) of Table 3.3-2.

18. Technical Specification 3/4.3.1 (Table 3.3-2)

Notation (##) of Table 3.3-2 states that the measured response time of the slowest RTD can be at least 13 seconds whereas the CE-STS and other CE plants with STS typically show 6 seconds. The applicant should provide information to clarify this issue. The information should provide, as a minimum, verification that the Technical Specification value for the RTD response time is consistent with that assumed in the safety analyses.

19. Technical Specification 3/4.3.1 (3.3-2)

The Palo Verde FSAR Section 7.2.1 (A) states that the Technical Specifications will include a discussion on the methods used to perform response time testing. However, no such discussion was found during the staff's proof and review of the Palo Verde Technical Specifications associated with the RPS or ESFAS. Therefore, the applicant should provide information to describe the methods to be implemented for response time testing of the various parameter channels associated with the RPS and ESFAS. If the information requested is already included in currently docketed material, please provide the appropriate reference(s).

20. Technical Specification 3/4.3.1 (Table 4.3-1)

From our review we noted an inconsistency between Tables 4.3-1 of the Palo Verde Technical Specifications and the CE STS. The

inconsistency is related to the portion of Notation (2) which discusses the adjustment of various parameters as associated with the calorimetric calculation. The applicant should correlate the Palo Verde notation with that of CE-STS Table 4.3-1 and provide a discussion justifying the inconsistency.

21. Technical Specification 3/4.3.2

The Palo Verde FSAR Section 7.3 identifies various BOP engineered safety features systems (ESFS). Review of the Palo Verde Technical Specifications revealed that the tables associated with Specification Section 3/4.3.2 do not include the actuation system instrumentation related to the subject BOP ESFS. The applicant should, therefore, revise the Section 3/4.3.2 tables to include appropriate requirements for the BOP engineered safety features actuation system signals. The Palo Verde SER (NUREG-0857) Section 7.3.1.1 (discussion on channel bypass) should be considered by the applicant when preparing applicable minimum channels operable requirements and associated Action statements.

22. Technical Specification 3/4.3.2 (Table 3.3-3)

The Palo Verde FSAR refers to CESSAR Section 7.3.1.1.10.7 for a discussion on the emergency feedwater system. As noted in the CESSAR SER (NUREG-0852) Section 7.3.2, the emergency feedwater actuation system (EFAS) employs, in part, AND logic circuitry for diverse

process variables whereas the engineered safety feature actuation system (ESFAS) associated with the remaining safety systems uses exclusively OR logic circuitry. Electrical interlocks to prevent the bypass of more than one channel of diverse variables is not provided and, thus, the potential exists whereby multiple bypasses could lead to a situation where the remaining operable emergency feedwater actuation channels may not meet the single failure criterion. The SER states that reference plants should adopt appropriate technical specifications which limit those multiple bypasses to preclude the situation described above. It is not clear from our review that the Palo Verde Technical Specifications contain appropriate restrictions for bypassing the multiple channels associated with the EFAS. The Technical Specifications associated with the EFAS currently reference the same action statements used for the safety systems utilizing only OR logic. The applicant is required to provide information to clarify this issue and to revise the Technical Specifications as necessary to comply with the subject SER requirement discussed above.

23. Technical Specification 3/4.3.2 (Table 3.3-3)

The Palo Verde FSAR references CESSAR Sections 7.2.1 and 7.3.1.1 for a description of the RPS and ESFAS. CESSAR Table 7.2-3 and Section 7.3.2.2.1 show that the RPS and ESFAS share the process measurement circuitry associated with the containment pressure and pressurizer pressure signals. Thus, it appears that Actions 13 and 14 of Technical Specification Table 3.3-3 should be revised to include

these process measurement circuits since they affect multiple functional units. Also, Technical Specification Table 3.3-1, Actions 2 and 3, should be revised to include containment pressure. The applicant should provide additional design details in this area if no Technical Specification revisions are proposed to resolve this apparent discrepancy.

24. Technical Specification 3/4.3.2 (Table 3.3-4)

The Palo Verde FSAR Section 7.3.1.2 references CESSAR FSAR Section 7.3.1.2 for setpoint values. The subject CESSAR FSAR section refers to Table 7.3-5. A comparison of CESSAR Table 7.3-5 with Technical Specification Table 3.3-4 shows that there is a discrepancy in the values. Accordingly, the staff recommends that the applicant provide justification to support the different values and revise the Technical Specifications or FSAR as necessary.

25. Technical Specification 3/4.3.3 (Subsection 3.3.3.5 and 3.3.3.6 respectively)

a. By memorandum dated December 30, 1982, from R. Mattson to D. Eisenhut, the Division of Systems Integration proposed technical specification changes to address the operability of the remote shutdown systems required under the provisions of GDC 19. These changes impose limiting conditions for operation and surveillance requirements on transfer switches, control circuits and both channels of monitoring instruments

for the remote shutdown system. We recommend that Section 3/4.3.3 of the Palo Verde Technical Specifications be modified in accordance with the December 30, 1982, memorandum.

b. By memorandum dated October 12, 1983, from R. Mattson to D. Eisenhut, the Division of Systems Integration proposed technical specification changes to address the operability of the post-accident monitoring instrumentation required under the provisions of NUREG-0737, Supplement #1. These changes update the Standard Technical Specifications to reflect the Regulatory Guide 1.97, Revision 2, graded approach to operability requirements depending on the importance to safety of the measurement of a specific variable. We recommend that Sections 3/4.3.3 and 6.8.4 of the Palo Verde Technical Specifications be modified in accordance with the October 12, 1983, memorandum.

26. <u>Technical Specification 3/4.4.8 (Subsections 3.4.8.3 and 4.4.8.3.1)</u>

Technical Specification Sections 3.4.8.3 and 4.4.8.3.1 should be revised to include appropriate requirements for the control room alarms incorporated to alert the operator that a low temperature overpressure event is in progress or to indicate that the shutdown cooling relief valves should be armed.

27 Technical Specification 3/4.5.1 and 3.4.7.13 (Subsections 4.5.1 and 4.7.13, respectively)

a. Review of the Palo Verde Technical Specifications revealed that the interlocks associated with the safety injection tank isolation valves and the shutdown cooling system inlet isolation valves do not have sufficient requirements to demonstrate operability. Specifically, the operability checks for the subject interlocks do not appear to include CHANNEL FUNCTIONAL TESTS or CHANNEL CALIBRATIONS. Accordingly, the staff recommends that the applicant provide appropriate surveillance requirements in the Palo Verde Technical Specifications.

b. Subsection 4.5.1 deviates from the CE-STS in that it does not include surveillance requirements for safety injection tank water level and pressure channels. The staff recommends that the applicant provide justification for this deviation or revise the Palo Verde Technical Specifications accordingly.

28 Technical Specification 3/4.5.2 (Subsection 4.5.2.e)

Item 3 of Technical Specification 4.5.2.e should be expanded to include the containment spray mini-flow valves and combined safety injection mini-flow return to refueling water tank valves as shown on Palo Verde FSAR Table 7.3-5.

29. Technical Specification 3/4.7.1 (Subsection 4.7.1.2)

Item a.1. of Specification 4.7.1.2 implies that Specification 4.0.4 is not applicable to the turbine-driven pump and motor-driven pumps for entry into MODE 3. The CE-STS specifies that the provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for only the turbine-driven pump. The applicant should provide information to justify this deviation from the CE-STS (refer to item C. of Specification 4.7.1.2).

30. Technical Specification 3/4.7.7 (Subsection 4.7.7)

Item d.2. of Specification 4.7.7 should be modified to clarify that the control room essential filtration system is to be verified operable for automatic start on a control room essential filtration actuation signal and on a safety injection signal. This issue is supported by the Palo Verde FSAR Section 7.3.1.1.10.10.

31. Technical Specification 3/4.7

Review of the Palo Verde Technical Specifications revealed that there are no specifications provided for the control room ventilation isolation system. The applicant should provide appropriate limiting conditions for operation and surveillance requirements in the Palo Verde Technical Specifications to address this system.

32. Technical Specification 3/4.9.9 (Subsection 4.9.9)

Technical Specification 4.9.9 should be revised to specify that containment purge valve isolation should be verified to occur on a containment isolation actuation signal (CIAS). Refer to the Palo Verde FSAR Section 7.3.1.1.10.9 for additional information related to this issue.

33. Technical Specification 3/4.9.12 (Subsection 3.9.12)

Review of the Palo Verde Technical Specification 3.9.12 revealed an inconsistency between it and the CE-STS. Action a. of the subject specification does not specify that fuel movement may proceed with one fuel handling building ventilation system inoperable if the operable system is in operation and discharging through at least one train of HEPA filters and charcoal absorbers. The applicant should provide information to justify the deviation or revise the Technical Specifications appropriately.

34. Technical Specification 3/4.3.2 (Table 4.3-2)

The applicant has been requested by letter dated July 2, 1984, to supply information related to testing at power of the engineered safety features actuation system (ESFAS). Specifically, the staff has requested the applicant to deliniate ESFAS actuation devices (i.e., subgroup relays) and associated actuated equipment (pumps, valves, etc.) that cannot be tested during power operation, and provide supporting justification. Other information was requested

which need not be detailed here. The applicant committed to supply the requested information (letter dated July 31, 1984, from E. E. Van Brunt, Jr., of APS to G. Knighton of NRC). The resolution of this issue will be reported at a later date upon receipt and review of the applicant's submittal.

35. Technical Specifications 2.2 and 3/4.3.2 (Tables 2.2-1 and 3.3-4)

As noted in the Palo Verde SER (NUREG-0852), the staff committed to audit the numerical values of setpoints for the protective systems. This audit was performed upon receipt of the Palo Verde Technical Specifications. Inconsistencies were found between setpoint values during the comparison of the Palo Verde and CESSAR FSAR information with that of the Technical Specifications as discussed in the various items listed above. The applicant is being requested to provide justification for such deviations. It should be noted that the Palo Verde FSAR references the CESSAR FSAR for a discussion on setpoints. The CESSAR FSAR (Sections 7.2.1.2, 7.2.2.1, and 7.3.2.1) states that the selection of trip setpoints is such that adequate protection is provided when all sensor and processing time delays and inaccuracies are taken into account. Actual uncertainties and delay times are obtained from calculations and tests performed on the protection systems and associated instrumentation.

Upon request, the applicant submitted (letter dated August 2, 1984, from E. E. Van Brunt, Jr., of APS to G. Knighton of NRC) a detailed discussion on the methodology used to establish the Technical Specification trip setpoints and allowable values for the reactor protection system and the engineered safety feature actuation system channels. Comparison of the August 2, 1984, submittal (Section 2.0) trip setpoint limits with those of the Technical Specifications revealed that the setpoint and allowable values are consistent with one another. However, the inconsistency described above related to the FSARs remains to be resolved with the applicant.

Further review of the August 2, 1984, submittal (Section 3.3) revealed an area of concern related to the calibration requirements for certain components associated with the various channels. It appears that there is an underlying assumption associated with the setpoint methodology that the instrument loop components contained within the plant protection system (PPS) cabinets are recalibrated monthly. This is because the setpoint methodology assumes the drift error allowance associated with PPS components to be based on the maximum expected drift over a 39-day period. Accordingly, it is recommended that the Palo Verde Technical Specifications be modified to include a requirement that the PPS cabinet components be recalibrated monthly. This can be accomplished by adding a footnote to the applicable tables similar in wording to the footnote that appears in the BWR/6-STS or the Grand Gulf Technical Specifications.

Please note that the review of the Palo Verde setpoint methodology is ongoing. You will be informed should any additional issues that impact the Technical Specifications develop.

ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix, Arizona 85036

received 9/19/84

JND

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

Dupe

~~8410020130~~ PDR

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt / RSK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc

cc: E. Licitra (w/a)
J. B. Martin (w/a)
R. Zimmerman (w/a)
G. Fiorelli (w/a)

PG 3/4 7-39 CHANGE: CEB

Surveillance 4.7.11.6.b. See page.

JUSTIFICATION:

Delete ... "and verifying that the hydrant barrel is dry "... This Spec is used for those plants in climates where freezing occurs. PVNGS, as discussed in the FSAR, is not subject to climates or weather that would cause water in the hydrant barrel from freezing and causing various damage.

PG 3/4 7-41 CHANGE: CEB

LCO and Surveillance 4.7.12.1.b. See page.

JUSTIFICATION:

Delete the fire windows reference. PVNGS doesn't have fire windows.

PG 3/4 7-42 CHANGE: CEB

Surveillance 4.7.12.2. See page.

JUSTIFICATION:

Deletion of Item A is justified in that PVNGS doesn't have any fire door supervision system as described; therefore, we cannot comply with this.

PG 3/4 7-43 CHANGE: PSB

Surveillance 4.7.13.b. See page.

JUSTIFICATION:

CE new numbers.

PG 3/4 8-1 CHANGE:
8-2
8-7

Tech. Spec. See pages. PSB ORAB

JUSTIFICATION:

The Tech. Spec. was changed to comply with NRC Generic letter 84-15.

Technical Specification 3/4.8.1, "A.C. Sources"

The proposed changes to Tech. Spec. 3/4.8.1 are a result of the applicable recommendations of NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability" dated July 2, 1984.

PSB
OLAB

A. Add action statement "B". The present Tech. Spec. 3/4.8.1.1 has a common action statement for an inoperable offsite circuit or an inoperable diesel generator. The proposed Tech. Spec. has separate action statements for an inoperable offsite circuit and an inoperable diesel generator.

B. Action statement "B" now requires the Additional Reliability Actions as prescribed in Table 4.8.2. As requested in Generic Letter 84-15, Table 4.8.2 is being implemented in the Technical Specifications as a means of establishing and verifying reliability goals for the diesel generators.

C. Revise the time requirement of Surveillance Requirement 4.8.1.1.2.a.4.

The action statements require a diesel generator start and load (Surveillance Requirement 4.8.1.1.2.a.4) with an inoperable A.C. power source. The time requirements of action statements a, b, c, and d are being revised as recommended in Generic Letter 84-15.

D. Delete Surveillance Requirement 4.8.1.1.3, which establishes reportibility requirements in accordance with Reg. Guide 1.108, Rev. 1 1977. Reportibility requirements are now established in Table 4.8-2. Generic Letter 84-15 states that Table 4.8.2 encompasses certain aspects of the existing requirements of Reg. Guide 1.108 and the qualitative recommendations of NUREG/CR-0660.

E. Delete existing Table 4.8-1 and replace with proposed Table 4.8-1, Table 4.8-2, and attachments to Table 4.8-2. This proposed change is recommended in Generic Letter 84-15.

PG 3/4 8-4 <u>CHANGE</u>:

4.8.1.1.2.d.2. See page. PSB

<u>DELETE</u>:

1) Train B auxiliary feedwater pump kw rating is incorrect. Should be 839 kw per FSAR Table 8.3-3, page 8.3-29 (attached).

2) Cannot meet single largest load requirement of 839 kw on Train A as HPSI pump is rates at 696 kw per FSAR Table 8.3-3, page 8.3-27 (attached).

8-5

PSB

PROOF AND REVIEW

PG 3/4 8-6 CHANGE:

4.8.1.1.2.d.12. See page.

JUSTIFICATION:

1) Through Performance Testing of 93SU-1SA01 and 93PE-1SA01, it was found that the General Atomic supplied sequencer will not meet the Tech. Spec. requirement of ±10% of the design interval between each load block which only allows a tolerance of 0.45 to 0.55 seconds for the HPSI pump.

2) Per the General Atomic Tech Manual J104-81, Section 4.2.8.2, Paragraph 2, Page 4-31 and SFR# 1SA-063 (attached), the tolerance should be ±1 second.

Recommended the following Tech. Spec. change to read:

"Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ±1 second of its design interval."

PG 3/4 8-12 CHANGE:

Table, Note A. See page.

JUSTIFICATION:

Note A requires that the float voltage be corrected for average electrolyte temperature. This correction factor is so small that it is insignificant; therefore, it can be eliminated.

PG 3/4 8-14 CHANGE:

LCO. See page.

JUSTIFICATION:

Delete the words in parentheses (both) and (between units at the some station). The Palo Verde electrical distribution system does not include breakers to tie the emergency busses on one unit to the emergency busses on another unit.

PG 3/4 8-18 CHANGE:

Surveillance 4.8.4.1.a.3. See page.

JUSTIFICATION:

PROOF AND REVIEW

PG 3/4 8-22 Typo's. → PSB

PG 3/4 8-23/24/25 Typo's. → PSB

PG 3/4 8-26 Typo's. → PSB

PG 3/4 8-27 CHANGE: → PSB

Table 3.8-2. Delete control panel CEDM M-G Set J-SFN-C02B.

JUSTIFICATION:

This can be deleted based on the fact that the individual CEAs penetrations are Tech Speced in Table 3.8-2. One of the other M-G sets was taken out in the pre proof/review copy of the Tech Specs. This really appears to be a typo.

PG 3/4 9-4 CHANGE: CSB

Surveillance 4.9.4.b.
... "Portions of Specification 4.9.9."

JUSTIFICATION:

The way the present Tech. Spec. is written it allows you to test the containment purge valves per applicable section of 4.6.3.2. This allows a potential for error in that one may not interpret the "Applicable" part of 4.6.3.2 the same as another. We believe that by specifying Spec 4.9.9 there is no allowance for error as in interpreting the applicable portion of a Tech. Spec. Spec 4.9.9 spells out what must be tested and the period for testing.

PG 3/4 9-6 CHANGE: ASB

See page.

JUSTIFICATION:

This Spec has been revised to use the correct terminology and limits associated with the PVNGS refueling machine.

CEA deletion is justified in that the major concern in this Tech Spec is damage to a fuel assembly. CEAs are not a problem compared to the damaged fuel assembly problem.

3/4 8-27a,b,c,d,e,f justification → PSB

See Page 35

PROOF

PG 3/4 8-18 DELETE

JUSTIFICATION

PVNGS uses Gould Shawmut fuses in our systems. PVNGS Tech Spec 4.8.4.1.a.3 requires functionally testing representative samples of such fuses. We request this Tech Spec to be deleted because the manufacturer states that removal and insertion of fuses located in clip type holders, fuse removal will destroy the fuse itself; removal and replacement of incline fuses may compromise cable integrity; removal of fuses that are crimped inline and wrapped with heat shrink insulation material will destroy the fuse itself; and due to seismic design of the holders, fuse removal will destroy the fuse itself.

Due to inaccessibility, destructive removal is required for testing the fuses. Gould Shawmut states that "under no condition can a current limiting fuse ever become less protective over life".

PROOF AND REVIEW

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Separate day fuel tanks with a minimum level of 2.75 feet (550 gallons of fuel), and

   2. A separate fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4 within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, 3, or 4*, the steam-driven auxiliary feed pump is OPERABLE.

   If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Until the steam generator is no longer required for heat removal.

a. With an offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a diesel generator of the above required A.C. electrical power sources inoperable,* demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours; restore diesel generators to OPERABLE status within (A**) days*** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. At the number of failures for the inoperable diesel indicated in Table 4.8-2 perform the Additional Reliability Actions prescribed in Table 4.8-2 and its attachments.

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter and Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With the diesel generator restored to OPERABLE status, follow Action Statement a. With the offsite circuit restored to OPERABLE status, follow Action Statement b.

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirement 4.8.1.1.2.4 ELECTRIC POWER SYSTEMS

**The maximum time that an individual diesel generator may be inoperable (A) shall be established by the licensee based on the manufacturer's recommendations and previous maintenance and repair experience. Every reasonable effort shall be made to restore individual diesel generators to operable status within that time period (A). Every reasonable effort shall be interpreted to mean that diagnosis and repairs are to begin immediately and are to continue uninterrupted until the diesel generator is declared operable or an orderly retreat to cold shutdown is initiated.

***The maximum total cumulative time that the diesel generators of the onsite emergency AC power system may be in the INOPERABLE status in a given year shall be proposed by the licensee.

PROOF AND REVIEW

ACTION: (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, follow Action Statement a.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With one diesel generator unit restored, follow Action Statement b and d.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 5500 kW in less than or equal to 120 seconds, and operates with a load greater than or equal to 5500 kW for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks.

c. At least once per 92 days and from new fuel prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:

1. A water and sediment content of less than or equal to 0.05 volume percent;

2. A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

3. A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.80 but less than or equal to 0.99 or an API gravity at 60°F of greater than or equal to 11 degrees but less than or equal to 47 degrees;

4. An impurity level of less than 2 mg of insolubles per 100 mL when tested in accordance with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and

PROOF AND REVIEW

5. The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a single largest load of greater than or equal to ~~885~~ 839 kW (Train B AFW pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz. (or 616 kw (Train A HPSI pump))

3. Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 6240 volts during and following the load rejection.

4. Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

PROOF AND REVIEW

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential, and low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6050 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 5500 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2d.6.b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 5500 kW.

9. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.
(Not Running)

10. Verifying that with the diesel generator operating in a test mode (connected to its bus); a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

(Running unloaded)

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± ~~10%~~ 1 second of its design interval.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) (turning gear engaged)
    b) (emergency stop)

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 600 rpm (steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz) in less than or equal to 10 seconds.

f. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or the equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission within 30 days in a Special Report pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

PROOF AND REVIEW

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last ~~100~~ 20 Valid Tests.*	Test Frequency
≤ 1	At least once per 31 days
~~2~~	~~At least once per 14 days~~
≥2 ~~3~~	At least once per 7 days **
~~≥ 4~~	~~At least once per 3 days~~

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the ~~last 100~~ tests ~~are~~ NUMBER OF AND FAILURES determined on a per ~~nuclear unit~~ DIESEL GENERATOR basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last ~~100~~ 20 valid tests". ~~Entry into this test schedule shall be made at the 31 day test frequency.~~

** THIS TEST FREQUENCY SHALL BE MAINTAINED UNTIL SEVEN CONSECUTIVE FAILURE FREE DEMANDS HAVE BEEN PERFORMED AND THE NUMBER OF FAILURES IN THE LAST 20 VALID DEMANDS HAS BEEN REDUCED TO ONE OR LESS

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PROOF AND REVIEW

TABLE 4.8-2

ADDITIONAL RELIABILITY ACTIONS

No. of failures in last 20 valid test	No of failures in last 100 valid tests	Action
3	6	Within 14 days prepare and maintain a report for NRC audit describing the diesel generator reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	Declare the diesel generator inoperable. Perform a requalification test program for the affected diesel generator. Requalification test program requirements are indicated in Attachment 2 to this table.

ATTACHMENT 1 TO TABLE 4.8-2

REPORTING REQUIREMENT

As a minimum the Reliability Improvement Program report for NRC audit shall include:

a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

b) analysis of failures and determination of root causes of failures

c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

e) the schedule for implementation of each action from d) above

f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

Once a licensee has prepared and maintain an initial report detailing the diesel generator reliability improvement program at his site, as defined above, the licensee need prepare only a supplemental report within 14 days after each failure during a valid demand for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

In addition to the above, submit a yearly data report on the diesel generator reliability.

ATTACHMENT 2 TO TABLE 4.8-2
DIESEL GENERATOR REQUALIFICATION PROGRAM

(1) Perform seven consecutive successful demands without a failure within 30 days of diesel generator being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of diesel generator of being restored to operable status.

(2) If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator of being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of being restored to operable status.

(3) If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive successful demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

(4) Following the second failure during the requalification test program, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

(5) During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

(a) The number of failures in the last 20 valid demands is less than 5.

(b) The number of failures in the last 100 valid demands is less than 11.

(c) In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exception will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above).

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above the provisions of those criteria alone will prevail.

3/4.8.2 D.C. SOURCES

OPERATNG

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the D.C. trains listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required D.C. trains inoperable, restore the inoperable D.C. trains to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required chargers inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 400 amperes for batteries A and B and 300 amperes for batteries C and D at 125 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 3.8-1

D.C. ELECTRICAL SOURCES

Train A

CHANNEL A	CHANNEL C
125V bus E-PKA-M41	125V D.C. bus E-PKC-M43
125V D.C. battery bank E-PKA-F11	125 V D.C. battery bank E-PKC-F13
Battery charger E-PKA-H11	Battery charger E-PKC-H13
or	or
Backup battery charger E-PKA-H15 (AC)	Backup battery charger E-PKA-H15 (AC)

Train B

CHANNEL B	CHANNEL D
125V D.C. bus E-PKB-M42	125V D.C. bus E-PKD-M44
125V D.C. battery bank E-PKB-F12	125V D.C. battery bank E-PKD-F14
Battery charger E-PKB-H12	Battery charger E-PKD-H14
or	or
Backup battery charger E-PKB-H16 (BD)	Backup battery charger E-PKB-H16 (BD)

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(a)	> 2.07 volts
Specific Gravity(b)		≥ 1.195	Not more than 0.020 below the average of all connected cells
	≥ 1.205(c)	Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(c)

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value, declare the battery inoperable.

(a) Corrected for average electrolyte temperature. e

(b) Corrected for electrolyte temperature and level.

(c) Or battery charging current is less than 2 amps when on charge.

PROOF AND REVIEW

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open both between redundant busses within the unit (and between units at the same station):

a. Train "A" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBA-S03
2. 480-volt ESF Load Center #E-PGA-L31
   a. MCC E-PHA-M31
3. 480-volt ESF Load Center #EPGA-L33
   a. MCC E-PHA-M33
   b. MCC E-PHA-M37
4. 480-volt ESF Load Center #E-PGA-L35
   a. MCC E-PHA-M35

b. Train "B" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBB-S04
2. 480-volt ESF Load Center #E-PGB-L32
   a. MCC E-PHB-M32
   b. MCC E-PHB-M38
3. 480-volt ESF Load Center #E-PGB-L34
   a. MCC E-PHB-M34
4. 480-volt ESF Load Center #E-PGB-L36
   a. MCC E-PHB-M36

c. 120-volt Channel A Vital A.C. Bus #E-PNA-D25 energized from its associated inverter connected to D.C. Channel A.*

d. 120-volt Channel B Vital A.C. Bus #E-PNB-D26 energized from its associated inverter connected to D.C. Channel B.*

e. 120-volt Channel C Vital A.C. Bus #E-PNC-D27 energized from its associated inverter connected to D.C. Channel C.*

f. 120-volt Channel D Vital A.C. Bus #E-PND-D28 energized from its associated inverter connected to D.C. Channel D.*

g. 125-volt D.C. Channel A energized from Battery Bank E-PKA-F11.

h. 125-volt D.C. Channel B energized from Battery Bank E-PKB-F12.

i. 125-volt D.C. Channel C energized from Battery Bank E-PKC-F13.

j. 125-volt D.C. Channel D energized from Battery Bank E-PKD-F14.

*Two inverters may be disconnected from their D.C. bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ESF Load Centers not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker aligmment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective devices shown in Table 3.8-2 inoperable:

a. Restore the protection device(s) to OPERABLE status or deenergize the circuits(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protection relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-2.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers' nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets it manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

PROOF AND REVIEW

TABLE 3.8-2

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1006	E-NHN-M1002B	SG WET LAYUP RECIRC. PUMP M-SGN-P01B
E-NHN-M1017	E-NHN-M1002B	CTMT/RADWASTE SUMP PUMP M-RDN-P03
E-NHN-M1003	E-NHN-M1002A	RCP 1B CONTROLLED BLEEDOFF VLV J-RCE-HV-431
E-NHN-M1004	E-NHN-M1002A	RCP 1B HP COOLER INLET VLV J-RCN-HV-447
E-NHN-M1005	E-NHN-M1002A	RCP 1B HP COOLER OUTLET VLV J-RCN-HV-451
E-NHN-M1010	E-NHN-M1002A	REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-M02B
E-NHN-M1014	E-NHN-M1002A	REACTOR CAVITY SUMP PUMP M-RDN-P01A
E-NHN-M2808	E-NHN-M2832C	RCP 2B CONTROL BLEEDOFF VLV J-RCE-HV-433
E-NHN-M2813	E-NHN-M2832C	RCP 2B HI PRESSURE COOLER INLET VLV J-RCN-HV-449
E-NHN-M1009	E-NHN-M1002A	RCP 2B HI PRESSURE COOLER OUTLET VLV J-RCN-HV-453
E-NHN-M1306	E-NHN-M1314A	SG 2 HOT LEG BLOWN ISO VLV J-SGE-HV-42
E-NHN-M1307	E-NHN-M1314A	SG 2 COLD LEG BLOWN ISO VLV J-SGE-HV-44
E-NHN-M1311	E-NHN-M1314D	WET LAY UP RECIRC PUMP M-SGN-P01A
E-NHN-M1316	E-NHN-M1314C	RCPT (30A) FOR SEAL CRANE ASSY MOTOR E-NHN-122A; E-NHN-122B
E-NHN-M1339	E-NHN-M1314C	MOVABLE INCORE DETECTOR DRIVE MACHINE M-RIN-M03A

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1321	E-NHN-M1344B	WELDING RCPT'S E-NHN-I107A B, C, D
E-NHN-M1331	E-NHN-M1314B	REACTOR CAVITY SUMP PUMP M-RDN-P01B
E-NHN-M1341	E-NHN-M1314B	REACTOR CAVITY FAN C DISCH DAMPER M-HCN-M02C
E-NHN-M1342	E-NHN-M1314B	CEDM ACU A INTAKE DAMPER M-HCN-M03A
E-NHN-M1343	E-NHN-M1314B	CEDM ACU B INTAKE DAMPER M-HCN-M03B
E-NHN-M1323	E-NHN-M1344A	REACTOR COOLANT OIL LIFT PUMP 2A M-RCN-P02C
E-NHN-M1332	E-NHN-M1344A	CTMT RADWASTE SUMP EAST M-RDN-P02
E-NHN-M1503	E-NHN-M1502A	RCP 1A CONTROL BLEEDOFF VLV J-RCE-HV-430
E-NHN-M1504	E-NHN-M1502A	RCP 2A CONTROL BLEEDOFF VLV J-RCE-HV-432
E-NHN-M1505	E-NHN-M1502A	RCP 1A HI PRESSURE COOLER INLET VLV J-RCN-HV-446
E-NHN-M1506	E-NHN-M1502A	RCP 2A HI PRESSURE COOLER INLET VLV J-RCN-HV-448
E-NHN-M1507	E-NHN-M1502A	RCP 1A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-450
E-NHN-M1511	E-NHN-M1535A	WELDING RCPT'S E-NHN-I12A, B, C
E-NHN-M1508	E-NHN-M1502B	RCP 2A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-452
E-NHN-M1509	E-NHN-M1502B	REACTOR CAVITY FAN A DISCH DAMPER M-HCN-M02A

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1533	E-NHN-M1502B	REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02D
E-NHN-M1534	E-NHN-M1535	CTMT BLDG MONO HOIST 1 TON M-ZCN-009
E-NHN-M1517	E-NHN-M1535	REACTOR COOLANT OIL LIFT PUMP M-RCN-P02A
E-NHN-M1902	E-NHN-M1917A	REACTOR CAVITY NORM CLG FAN M-HCN-A03A
E-NHN-M1904	E-NHN-M1917B	REACTOR CAVITY NORM CLG FAN M-HCN-A03C
E-NHN-M1907	E-NHN-M1917	CEDM NORM ACU-A HEXCH OUTLET VLV J-NCN-HV-485
E-NHN-M1911	E-NHN-M1917	CTMT NORM ACU-C CHILLED WTR INLET VLV J-WCN-HV-59
E-NHN-M1912	E-NHN-M1917	CTMT NORM ACU-A CHILLED WTR INLET VLV J-WCN-HV-57
E-NHN-M2008	E-NHN-M2010	CEDM NORM ACU-B HEXCH OUTLET VLV J-NCN-HV-486
E-NHN-M2003	E-NHN-M2010	CTMT NORM ACU-B CHILL WATER INLET VLV J-WCN-HV-58
E-NHN-M2004	E-NHN-M2010	CTMT NORM ACU-D CHILL WATER INLET VLV J-WCN-HV-60
E-NHN-M2006	E-NHN-M2010A	REACTOR CAVITY NORM CLG FAN M-HCN-A03B
E-NHN-M2007	E-NHN-M2016	REACTOR CAVITY NORM CLG FAN M-HCN-A03D
E-NHN-M2803	E-NHN-M2827A	CEDM ACU C INTAKE DAMPER M-HCN-M03C
E-NHN-M2804	E-NHN-M2827A	CEDM ACU D INTAKE DAMPER M-HCN-M03D

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M2805	E-NHN-M2827A	SG1 COLD LEG BLOWDOWN ISO VLV J-SGE-HV-91
E-NHN-M2806	E-NHN-M2827~~A~~ B	SG HOT LEG BLOWDOWN ISOLATION VALVE J-SGE-HV43
E-NHN-M2827	E-NHN-M2827~~A~~ ~~B~~ A	REACTOR COOL PUMP OIL LIFT PUMP 1B M-RCN-P02B
E-NHN-M2828	E-NHN-M2827~~B~~ A	REACTOR COOLANT PUMP OIL LIFT PUMP 2B M-RCN-P~~2~~02D
E-NHN-M2809	E-NHN-M2827C	CONTAINMENT EQUIP HATCH J-ZCN-E02
E-NHN-M2811	E-NHN-M2832A	30A RECEPTACLES FOR CTMT BLDG JIB CRANE
E-NHN-M2818	E-NHN-M2832A	30A RECEPTACLES FOR SEAL CRANE ASSY MOT
E-NHN-M2817	E-NHN-M2832B	CTMT BLDG MONORAIL HOIST 1 TON M-ZCN-G03
E-NHN-M2819	E-NHN-M2832B	30A RECEPTACLES FOR CTMT BLDG JIB CRANE G04 A, B
E-NHN-M2820	E-NHN-M2832D	CTMT BLDG ELEV #2 CONTROLLER J-ZCN-E01
E-NHN-M2821	E-NHN-M2828C	MULTIPLE STUD TENSIONER M-ZCN-M15
E-NHN-M2822	E-NHN-M2828B	WELDING RECPTS E-NHN-I09 B, C, D
E-NHN-M2801A	E-NHN-M2827B	FUEL TRANSFER SYS CONTROL CONSOLE E-PCE-D02
E-NHN-M2833	E-NHN-M2827B	REFUELING MACHINE E-PCE-J02
E-NHN-M2833A	E-NHN-M2827B	CEA CHANGE PLATFORM E-PCE-J01

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PGB-L34D2	E-NGN-B34D2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A01D
E-PGB-L34D3	E-NGN-B34D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02D
E-PGB-L36D3	E-NGN-~~B63D3~~ B36D3 (FUSE)	CTMT NOR ACU FAN M-HCN-A01B
E-PHA-M3318	E-PHA-M3334	SAFETY INJECT TANK 4 ISOL VLV J-SIA-UV-644
E-PHA-M3316	E-PHA-M3316A	SAFETY INJECT TANK 3 ISOL VLV J-SIA-UV-634
E-PHB-M3404	E-PHB-M3405B	NCWS RET INT CTMT ISOL.VLV J-NCB-UV-403
E-PHA-M3519	E-PHA-M3521A	CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-48
E-PHA-M3521	E-PHA-M3517	CTMT PRG RFL MODE ISO VLV J-CPA-UV-2B
E-PHA-M3503	E-PHA-M3507A	SHUT DN CLG ISOL LOOP 1 VLV J-SIA-UV-651
E-PHA-M3508	E-PHA-M3511A	CTMT/RAD SUMP CTMT INT ISO VLV J-RDA-UV-23
E-PHA-M3512	E-PHA-M3513A	CTMT SUMP ISOL TRAIN A VLV J-SIA-UV-673
E-PHB-M3622	E-PHB-M3629	CTMT PRG REFULING MODE ISO VLV J-CPB-UV-3A
E-PHB-M3604	E-PHB-M3604A	SHUT DN CLG ISOL LOOP 2 VLV J-SIB-UV-652
E-PHB-M3619	E-PHB-M3641A	SAFETY INJECTION TANK ISOL VLV J-SIB-UV-614
E-PHB-M3624	E-PHA-M3607A	CTMT PRG PWR ACCESS MODE ISO VLV J-CPB-UV-5A

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PHB-M3613	E-PHB-M3613A	CTMT SUMP ISOL TRAIN B VLV J-SIB-UV-675
E-PHB-M3618	E-PHB-M3641	SAFETY INJECTION TANK 2 ISO VLV J-SIB-UV-624
E-PHA-M3704	E-PHA-M3703A	WASTE GAS HEADER CONT. ISOLATION VALVE GRA-UV1
E-PHA-M3715	E-PHA-M3719	H_2 CONT TRAIN A UPSTM SUP ISO VLV J-HPA-UV-1
E-PHB-M3816	E-PHB-M3836	H_2 CTMT TRAIN B UPSTM SUP ISO VLV J-HPB-UV-2
E-PHB-M3811	E-PHB-M3813A	NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61
E-PKD-B44	E-PKD-M4411	SHUTDOWN CLG ISOL VLV J-SID-UV-654
E-PKC-B43	E-PKC-M4311	SHUTDOWN COOLING ISOL VLV J-SIC-UV-653
E-NNN-D1113	E-NNN-D11	MOVABLE INCORE DRIVE SYS #I 800VA, M-RIN-M03A VIA E-RIN-J01A
E-NNN-D1213	E-NNN-D12	MOVABLE INCORE DRIVE SYS #II 800VA, M-RIN-M03B VIA E-RIN-J01A
E-NNN-D1526	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E02
E-NNN-D1525	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E01
E-NNN-D1626	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E04
E-NNN-D1625	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E03
E-QAN-B02	E-QAN-D05 CKT 2, 4, 6	LIGHTING PANEL E-QAN-D05B CTMT BLDG EL 100'

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-QAN-B03	E-QAN-D05 CKT 7, 9, 11	LIGHTING PANEL E-QAN-D05C CTMT BLDG EL 100'
E-QAN-B04	E-QAN-D05 CKT 8, 10, 12	LIGHTING PANEL E-QAN-D05D CTMT BLDG EL 140'
E-QAN-B05	E-QAN-D05 CKT 19, 21, 23	LIGHTING PANEL E-QAN-D05F CTMT BLDG EL 140'
E-QAN-B06	E-QAN-D05 CKT 13, 15, 17	LIGHTING PANEL E-QAN-D05E CTMT BLDG EL 140'
E-QBN-B01	E-QBN-D91 CKT 19, 21, 23	LIGHTING PANEL E-QBN-D73A CTMT BLDG EL 100'
E-QBN-B02	E-QBN-D91 CKT 20, 22, 24	LIGHTING PANEL E-QBN-D73B CTMT BLDG EL 140'
E-NHN-D1514	E-NHN-M1526	TO OPERATION CAMERA JB# 22
E-RCN-D0101	E-NGN-L11C2	PZR BU HTR M-RCE-B07, B13, A01
E-NAN-D2614	E-NHN-M2618	TO OPERATION CAMERA JB# 21
E-RCN-D0102	E-NGN-L11C2	PZR BU HTR M-RCE-B03, A09, A15
E-RCN-D0302	E-NGN-L11C3	PZR BU HTR M-RCE-B04, A11, A16
E-RCN-D0301	E-NGN-L11C3	PZR BU HTR M-RCE-A02, A07, A13
E-RCN-D0202	E-NGN-L12C2	PZR BU HTR M-RCE-B06, B12, A18
E-RCN-D0201	E-NGN-L12C2	PZR BU HTR M-RCE-B16, A04, A08
E-RCN-D0402	E-NGN-L12C3	PZR BU HTR M-RCE-B15, A03, A10
E-RCN-D0401	E-NGN-L12C3	PZR BU HTR M-RCE-A17, A06, A12

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NAN-S01M	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01A (C.E. NO. 1A)
E-NAN-S01L	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01C (C.E. NO. 2A)
E-NAN-S02L	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01B (C.E. NO. 1B)
E-NAN-S02-M	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01D (C.E. NO. 2B)
E-NGN-L03C2	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01C
E-NGN-L03C3	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01D
E-NGN-L03D2	FUSE IN BKR.	CTMT POLAR CRANE M-ZCN-G01
E-NGN-L06C2	E-NGN-B06C2 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01A
E-NGN-L09C4	E-NGN-B09C4 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01B
E-NGN-L10C2	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN-E01A
E-NGN-L10C3	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN-E01B
E-NGN-L10C4	FUSE IN BKR.	CONTROL PANEL CEDM M-G SET J-SFN-C02B
E-NGN-L11C4	E-NGN-L11B2	PROPORTIONAL HTR BANK M-RCE-B2, B8, B14
E-NGN-L12C4	E-NGN-L12B2	PROPORTIONAL HTR BANK M-RCE-B5, B11, B17
CEA 06 CB101	F101, F102, F103	CEA 06
CEA 08 CB102	F104, F105, F106	CEA 08
CEA 10 CB103	F107, F108, F109	CEA 10

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PROOF AND RE

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M1341 (FUSE)	E-NHN-M1341 (FUSE)	REACTOR CAVITY FAN C DISCH DAMPER M-HCN-M02C
E-NHN-M1306 (FUSE)	E-NHN-M1306 (FUSE)	SG 2 HOT LEG BLDWN ISOL VLV J-SGE-HV-42
E-NHN-M1307 (FUSE)	E-NHN-M1307 (FUSE)	SG2 COLD LEG BLDWN ISO VLV J-SGE-HV-44
E-NHN-M2803 (FUSE)	E-NHN-M2803 (FUSE)	CEDM ACU C INTAKE DAMPER M-HCN-M03C
E-NHN-M2804 (FUSE)	E-NHN-M2804 (FUSE)	CEDM ACU D INTAKE DAMPER M-HCN-M03D
E-NHN-M2805 (FUSE)	E-NHN-M2805 (FUSE)	SG1 COLD LEG BLDWN ISO VLV J-SGE-HV-41
E-NHN-M2806 (FUSE)	E-NHN-M2806 (FUSE)	SG 1 HOT LEG BLDWN ISO VLV J-SGE-HV-43
E-NHN-M2808 (FUSE)	E-NHN-M2808 (FUSE)	RCP 2B CONTROL BLEEDOFF VALVE J-RCE-HV-433
NHN-M2813 FUSE)	E-NHN-M2813 (FUSE)	RCP 2B HI PRESS COOLER INLET VLV J-RCN-HV-44
E-NHN-M1004 (FUSE)	E-NHN-M1004 (FUSE)	RCP 1B HP COOLER INLET VLV J-RCN-HV-447
E-NHN-M1006 (FUSE)	E-NHN-M1006 (FUSE)	SG WET LAYUP RECIRC PUMP M-SGN-P01B
E-NHN-M1005 (FUSE)	E-NHN-M1005 (FUSE)	RCP 1B HP COOLER INLET VLV J-RCN-HV-451
E-NHN-M1009 (FUSE)	E-NHN-M1009 (FUSE)	RCP 2B HI PRESS COOLER INLET VLV J-RCN-HV-45
E-NHN-M1010 (FUSE)	E-NHN-M1010 (FUSE)	REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-MO
E-PHA-M3604 (FUSE)	E-PHA-M3604 (FUSE)	SHUT DN CLG ISO LOOP 2 VLV J-SIB-UV-65
E-PHA-M3613 (FUSE)	E-PHA-M3613 (FUSE)	CTMT SUMP ISOL TRAIN-B VLV J-SIB-UV-675
PHA-M3619 -SE)	E-PHA-M3619 (FUSE)	SAFETY INJECTION TANK 1 ISOL VLV J-SIB-UV-614
NHN-M1904 (FUSE)	E-NHN-M1904 (FUSE)	REACTOR CAVITY NORM CLG FAN M-HCN-A03C

NOTE: THE ABOVE BACKUP PROTECTION FUSES WERE ADDED BY DCP NO. 1CE-[illegible]-036, 2CE-PH-036, 3CE-PH-036.

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M1907 (FUSE)	E-NHN-M1907 (FUSE)	CEDM NORM. ACU A HEXCH OUTLET VLV J-NCN-HV-45
E-NHN-M1911 (FUSE)	E-NHN-M1911 (FUSE)	CTMT NORM ACU-C CHILLED WATER INLET VLV J-WCN-HV
E-NHN-M1912 (FUSES)	E-NHNM1912 (FUSE)	CTMT NORM ACU-A CHILLED WATER INLET VLV J-WCN-HV-
E-NHN-M2003 (FUSE)	E-NHN-M2003 (FUSE)	CTMT NORM ACU-B CHILLED WATER INLET VLV J-WCN-HV-5
E-NHN-M2004 (FUSE)	E-NHN-M2004 (FUSE)	CTMT NORM ACU-D CHILLED WATER. INLET VLV J-WCN-HV-
E-NHN-M2006 (FUSE)	E-NHN-M2006 (FUSE)	REACTOR CAVITY NORM CLG FAN M-HCN-A03B
E-NHN-M2008 (FUSE)	E-NHN-M2008 (FUSE)	CEDM NORM ACU B HEX OUTLET VLV J-HCN-HV-4
E-NHN-M1503 (FUSE)	E-NHN-M1503 (FUSE)	RCP 1A CONTROL BLEEDOFF V J-RCE-HV-430
E-NHN-M1504 (FUSE)	E-NHN-M1504 (FUSE)	RCP 2A CONTROL BLEEDOFF VL J-RCE-HV-432
E-NHN-M1505 (FUSE)	E-NHN-M1505 (FUSE)	RCP 1A HI PRESS COOLER INLET VLV J-RCN-HV-446
E-NHN-M1506 (FUSE)	E-NHN-M1506 (FUSE)	RCP 2A HI PRESS COOLER INLET VLV J-RCN-HV-445
E-NHN-M1507 (FUSE)	E-NHN-M1507 (FUSE)	RCP 1A HI PRESS COOLER OUTLET VLV J-RCN-HV-450
E-NHN-M1508 (FUSE)	E-NHN-M1508 (FUSE)	RCP 2A HI PRESS COOLER OUTLET VLV J-RCN-HV-452
E-NHN-M1509 (FUSE)	E-NHN-M1509 (FUSE)	REACTOR CAVITY FAN A DISC DAMPER M-HCN-M02A
E-NHN-M1533 (FUSE)	E-NHN-M1533 (FUSE)	REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02
E-PHB-M3811 (FUSE)	E-PHB-M3811 (FUSE)	NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61
E-PHB-M3816 (FUSE)	E-PHB-M3816 (FUSE)	H2 CTMT TR B UPSTM SUP. ISO VLV J-HPB-UV-2
E-NHN-M7102 (FUSE)	E-NHN-M7102 (FUSE)	CTMT NORM ACU A DISCH DAMPER M-HCN-M01A

NOTE: THE ABOVE SHORT PROTECTIVE FUSES WERE ADDED BY [illegible] NO. 1SE-PH-036, 2SE-PH-036, 3SE [illegible] -036

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES — CONTROL CIRCUITS

[illegible]MARY DEVICE [illegible]UMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M7103 (FUSE)	E-NHN-M7103 (FUSE)	CTMT NORM ACU C DISCH DAMPER M-HCN-M01C
M-HCN-E01A (FUSE)	M-HCN-E01A (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01A
M-HCN-E01B (FUSE)	M-HCN-E01B (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01[illegible]
M-HCN-E01C (FUSE)	M-HCN-E01C (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01[illegible]
M-HCN-E01D (FUSE)	M-HCN-E01D (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01[illegible]

NOTE: THE ABOVE BACKUP PROTECTION FUSES WERE ADDED B[illegible] DCP NO. 1SE-PH-036, 2CE-PH-036, 3CE-PH-036

E-ZAA-C03 (FUSE)	E-PKA-D2109	REACTOR DRAIN TANK OUTLE[illegible] ISOL VLV J-CHAUV-560
E-ZAA-C03 ([illegible]E)	E-PKA-D2109	SI TK RWT RTN HDR CTMT ISO VLV J-SIA-UV-682
[illegible]AA-C03 ([illegible]SE)	E-PKA-D2109	REGENERATIVE HEAT EXCH T[illegible] AUX SPRAY VLV J-CHA-HV-2[illegible]
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VLV J-SSA-UV-203
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VL[illegible] J-SSA-UV-204
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VL[illegible] J-SSA-UV-205
E-ZAA-C04 (FUSE)	E-PKA-D2102	PRESSURIZER VENT VALVE J-RCA-HV-103
E-ZAA-C05 (FUSE)	E-PKA-D2114	STEAM GEN BLOWDOWN CTMT IS[illegible] VLV J-SGA-UV-500P
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO V[illegible] J-SGA-UV-204
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO VL[illegible] J-SGA-UV-211
[illegible]AA-C05 ([illegible]SES)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO VL[illegible] J-SGA-UV-220

[illegible]E: THE ABOVE BACKUP PROTECTION CONTROL FUSES WERE ADDED BY DCP NO. 1SE-PK-[illegible], 2[illegible]-PK-0[illegible] 3C[illegible]-PK-02[illegible]

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR REVIEW

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	SECONDARY DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TK NITROGEN SP VLV J-SIA-HV-619
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TK NITROGEN SPL VLV J-SIA-HV-629
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALV J-SIA-HV-605
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-606
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VLV J-SIA-HV-607
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VLV J-SIA-HV-608
E-ZAA-C06 (FUSE)	E-PKA-D2121	RC SYSTEM VENT TO CTMT VL J-RCA-HV-106
E-ZAB-C03 (FUSE)	E-PKB-D2209	REGEN HEAT EXCH. TO AUX SPRAY VALVE J-CHB-HV-203
E-ZAB-C03 (FUSE)	E-PKB-D2209	REACTOR COOLANT VENT VLV J-RCB-HV-102
E-ZAB-C03 (FUSE)	E-PKB-D2209	SAFETY INJ TK FILL + DRAIN VLV J-SIB-UV-611
E-ZAB-C03 (FUSE)	E-PKB-D2209	SI TK CHECK VLV LEAKAGE LINE 150 VLV J-SIB-UV-618
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADN MONITORING IS VLV J-HCB-UV-44
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADN MONITORING IS VLV J-HCB-UV-47
E-ZAB-C04 (FUSE)	E-PKB-D2202	REACTOR COOLANT VENT VLV J-RCB-HV-108
E-ZAB-C04 (FUSE)	E-PKB-D2202	SAFETY INJ TK FILL + DRAIN VLV J-SIB-UV-621
E-ZAB-C04 (FUSE)	E-PKB-D2202	SI TK CHECK VLV LEAKAGE LINE 150 VLV J-SIB-UV-625
E-ZAB-C05 (FUSE)	E-PKB-D2214	REACTOR COOLANT VENT VLV J-RCB-HV-109
E-ZAB-C05 (FUSE)	E-PKB-D2214	STEAM GEN BLOWDOWN CTMT ISO VLV J-SGB-UV-500R

NOTE: THE ABOVE BACKUP PROTECTION CONTROL CIRCUIT FUSES WERE [illegible]

ADDED BY DCP [illegible] E-PK-D2[illegible]

TABLE 3.3-2 (CONTINUED)

PROOF AND REVIEW

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

P[illegible]ARY DEVICE [illegible]IBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO V[illegible] J-SGB-UV-222
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO V[illegible] J-SGB-UV-224
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO V[illegible] J-SGB-UV-226
E-ZAB-C06 (FUSE)	E-PKB-D2221	REACTOR COOLANT VENT VLV J-RCB-HV-105
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK NITROGEN SPL[illegible] VLV J-SIB-UV-612
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK NITROGEN SPLY VLV J-SIB-UV-622
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-613
E-ZAB-C06 ([illegible]USE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-623
[illegible]AB-C06 ([illegible]SE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-633
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VL[illegible] J-SIB-HV-643
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TK NITROGEN SP[illegible] VLV J-SIA-HV-639
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TK NITROGEN SPL[illegible] VLV J-SIA-HV-649
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO RD[illegible] VALVE J-CHA-HV-507
E-ZJA-C03 (FUSE)	E-PKA-D2111	LETDOWN LINE TO REGEN HEAT EXCH CTMT ISO VALVE J-CHA-UV-51[illegible]
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO VCT VALVE J-CHA-UV-506
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TK FILL & DRAIN VLV J-SIB-UV-641
[illegible]JB-C01 ([illegible]E)	E-PKB-D2201	SI TK CHECK VLV LEAKAGE LI[illegible] ISO VALVE J-SIB-UV-649
E-ZJB-C01 (FUSE)	E-PKB-D2201	HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-322

[illegible] ... DCP NO. [illegible]

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

[illegible]IARY DEVICE [illegible]MBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TK NITROGEN SPLY VLV J-SIB-HV-632
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ KT NITROGEN SPLY VLV J-SIB-HV-692
E-ZJB-C03 (FUSE)	E-PKB-D2211	LETDOWN LINE TO REGEN HEAT EXCH VALVE J-CHB-UV-515
E-ZJB-C03 (FUSE)	E-PKB-D2211	SAFETY INJ TK FILL & DRAIN VL J-SIB-UV-631
E-ZJB-C03 (FUSE)	E-PKB-D2211	SI TK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-639
E-ZJB-C03 (FUSE)	E-PKB-D2211	HOT LEG INJECT CHECK VLV LEAK ISO VLV J-SIB-UV-332
E-ZAN-C01 (FUSE)	E-NKN-D4226	SEAL INJECT VALVES TO RCP J-CHE-FV-241
E-ZAN-C01 ([illegible]SE)	E-NKN-D4224	SEAL INJECT VALVES TO RCP J-CHE-FV-242
[illegible]AN-C01 ([illegible]SE)	E-NKN-D4222	SEAL INJECT VALVES TO RCP J-CHE-FV-244
E-ZAN-C01 (FUSE)	E-NKN-D4224	POST ACDT SMPLG SYS ISO VLV J-CHN-HV-923
E-ZAN-C01 (FUSE)	E-NKN-D4224	REACTOR VESSEL SEAL DRAIN TO RDT VLV J-RCE-HV-403
E-ZAN-C01 (FUSE)	E-NKN-D4224	SI DRAIN TO REACTOR DRAIN T VLV J-SIE-HV-661
E-ZAN-C02 (FUSE)	E-NKN-D4216	SEAL INJECT VALVES TO RC J-CHE-FV-243
E-ZAN-C02 (FUSE)	E-NKN-D4216	REGEN HEAT EXCH TO CHARGING LINE VLV J-CHE-PDV-240
E-PGB-L32E2 (FUSE)	E-PGB-L32E2 (FUSE	CEDM NORM ACU FAN B M-HCN-A02B
E-PGB-L34D2 (FUSE)	E-PGB-L34D2 (FUSE)	CTMT NORM ACU FAN D M-HCN-A01D

[illegible]: THE ABOVE BACKUP PROTECTION CONTROL CIRCUIT FUSES WERE ADDED BY [illegible] [illegible]-PK-022, [illegible]-PK-02[illegible], [illegible]-PK-02[illegible]

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 12 CB104	F110, F111, F112	CEA 12
CEA 07 CB101	F101, F102, F103	CEA 07
CEA 09 CB102	F104, F105, F106	CEA 09
CEA 11 CB103	F107, F108, F109	CEA 11
CEA 13 CB104	F110, F111, F112	CEA 13
CEA 74 CB101	F101, F102, F103	CEA 74
CEA 76 CB102	F104, F105, F106	CEA 76
CEA 78 CB103	F107, F108, F109	CEA 78
CEA 80 CB104	F110, F111, F112	CEA 80
CEA 75 CB101	F101, F102, F103	CEA 75
CEA 77 CB102	F104, F105, F106	CEA 77
CEA 79 CB103	F107, F108, F109	CEA 79
CEA 81 CB104	F110, F111, F112	CEA 81
CEA 22 CB101	F101, F102, F103	CEA 22
CEA 24 CB102	F104, F105, F106	CEA 24
CEA 26 CB103	F107, F108, F109	CEA 26
CEA 28 CB104	F110, F111, F112	CEA 28
CEA 23 CB101	F101, F102, F103	CEA 23
CEA 25 CB102	F104, F105, F106	CEA 25

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 27 CB103	F107, F108, F109	CEA 27
CEA 29 CB104	F110, F111, F112	CEA 29
CEA 34 CB101	F101, F102, F103	CEA 34
CEA 36 CB102	F104, F105, F106	CEA 36
CEA 38 CB103	F107, F108, F109	CEA 38
CEA 40 CB104	F110, F111, F112	CEA 40
CEA 35 CB101	F101, F102, F103	CEA 35
CEA 37 CB102	F104, F105, F106	CEA 37
CEA 39 CB103	F107, F108, F109	CEA 39
CEA 41 CB104	F110, F111, F112	CEA 41
CEA 55 CB101	F101, F102, F103	CEA 55
CEA 58 CB102	F104, F105, F106	CEA 58
CEA 61 CB103	F107, F108, F109	CEA 61
CEA 64 CB104	F110, F111, F112	CEA 64
CEA 54 CB101	F101, F102, F103	CEA 54
CEA 57 CB102	F104, F105, F106	CEA 57
CEA 60 CB103	F107, F108, F109	CEA 60
CEA 63 CB104	F110, F111, F112	CEA 63
CEA 56 CB101	F101, F102, F103	CEA 56
CEA 59 CB102	F104, F105, F106	CEA 59

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 62 CB103	F107, F108, F109	CEA 62
CEA 65 CB104	F110, F111, F112	CEA 65
CEA 66 CB101	F101, F102, F103	CEA 66
CEA 68 CB102	F104, F105, F106	CEA 68
CEA 70 CB103	F107, F108, F109	CEA 70
CEA 72 CB104	F110, F111, F112	CEA 72
CEA 67 CB101	F101, F102, F103	CEA 67
CEA 69 CB102	F104, F105, F106	CEA 69
CEA 71 CB103	F107, F108, F109	CEA 71
CEA 73 CB104	F110, F111, F112	CEA 73
CEA 02 CB101	F101, F102, F103	CEA 02
CEA 03 CB102	F104, F105, F106	CEA 03
CEA 04 CB103	F107, F108, F109	CEA 04
CEA 05 CB104	F110, F111, F112	CEA 05
CEA 42 CB101	F101, F102, F103	CEA 42
CEA 43 CB102	F104, F105, F106	CEA 43
CEA 44 CB103	F107, F108, F109	CEA 44
CEA 45 CB104	F110, F111, F112	CEA 45
CEA 82 CB101	F101, F102, F103	CEA 82
CEA 83 CB102	F104, F105, F106	CEA 83

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 84 CB103	F107, F108, F109	CEA 84
CEA 85 CB104	F110, F111, F112	CEA 85
CEA 18 CB101	F101, F102, F103	CEA 18
CEA 19 CB102	F104, F105, F106	CEA 19
CEA 20 CB103	F107, F108, F109	CEA 20
CEA 21 CB104	F110, F111, F112	CEA 21
CEA 86 CB101	F101, F102, F103	CEA 86
CEA 87 CB102	F104, F105, F106	CEA 87
CEA 88 CB103	F107, F108, F109	CEA 88
CEA 89 CB104	F110, F111, F112	CEA 89
CEA 14 CB101	F101, F102, F103	CEA 14
CEA 15 CB102	F104, F105, F106	CEA 15
CEA 16 CB103	F107, F108, F109	CEA 16
CEA 17 CB104	F110, F111, F112	CEA 17
CEA 46 CB101	F101, F102, F103	CEA 46
CEA 48 CB102	F104, F105, F106	CEA 48
CEA 50 CB103	F107, F108, F109	CEA 50
CEA 52 CB104	F110, F111, F112	CEA 52
CEA 47 CB101	F101, F102, F103	CEA 47
CEA 49 CB102	F104, F105, F106	CEA 49

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 51 CB103	F107, F108, F109	CEA 51
CEA 53 CB104	F110, F111, F112	CEA 53
CEA 30 CB101	F101, F102, F103	CEA 30
CEA 31 CB102	F104, F105, F106	CEA 31
CEA 32 CB103	F107, F108, F109	CEA 32
CEA 33 CB104	F110, F111, F112	CEA 33
CEA 01 CB101	F101, F102, F103	CEA 01

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PHA-D33-03	E-PHA-M3333	INDICATING LIGHTS FOR VLV J-SIA-UV-634
E-PHA-D33-04	E-PHA-M3333	INDICATING LIGHTS FOR VLV J-SIA-644
E-PHB-D36-01	E-PHB-M3639	INDICATING LIGHTS FOR VLV J-SIB-UV-614
E-PHB-D36-02	E-PHB-M3639	INDICATING LIGHTS FOR VLV J-SIB-UV-624
E-NHN-D28-04	E-NHN-M2831	CONTAINMENT PREACCESS NORMAL AFU MOTOR SPACE HEATER FOR M-MCN-F01AH
E-NHN-D28-14	E-NHN-M2831	FLOW SWITCH J-HCN-FSL-29 FOR DUCT HEATERS M-HCN-E01A AND B
E-NHN-D28-16	E-NHN-M2831	CONTAINMENT AHU DUCT HEATERS M-HCN-E01A AND B TEMPERATURE CONTROL J-HCN-TC-29

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D28-18	E-NHN-M2831	FLOW SWITCH J-HCN-FSL-31 FOR DUCT HEATERS M-HCN-E01C AND D
E-NHN-D13-04	E-NHN-M1330	CONTAINMENT ACU DUCT HEATERS M-HCN-E01C AND D TEMPERATURE CONTROLLER J-HCN-TC-31
[illegible]NHN-D13-22	E-NHN-M1330	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01AH
E-NHN-D15-01	E-NHN-M1527	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01BH
E-NHN-D15-02	E-NHN-M1527	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01D[illegible]
E-NHN-D15-06	E-NHN-M1527	CONTAINMENT PREACCESS NORMAL AFU FAN MOTOR SPACE HEATER M-HCL-F[illegible]
[illegible]E-NHN-D10-01	E-NHN-M1028	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01AH

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACK UP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D10-02	E-NHN-M1028	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01CH
E-NHN-D10-20	E-NHN-M1028	STEAM GENERATOR WET LAY UP PUMP MOTOR SPACE HEATER M-SGN-P01B
E-NHN-D19-05	E-NHN-M1915	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02AH
E-NHN-D19-06	E-NHN-M1915	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02CH
E-NHN-D19-07	E-NHN-M1915	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01AH
E-NHN-D19-08	E-NHN-M1915	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01CH
E-NHN-D19-10	E-NHN-M1915	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03AH

TABLE 3.8-2 - (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTION DEVICES.

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D19-12	E-NHN-M1915	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03CH
E-NHN-D20-05	E-NHN-M2014	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02BH
E-NHN-D20-06	E-NHN-M2014	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02DH
E-NHN-D20-07	E-NHN-M2014	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01D
E-NHN-D20-08	E-NHN-M2014	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01BH
E-NHN-D20-10	E-NHN-M2014	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03BH
E-NHN-D20-12	E-NHN-M2014	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03DH

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day tank with a minimum level of 2.75 feet (550 gallons of fuel),

   2. A fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to ( ) square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for Requirement 4.8.1.1.2a.5.) and 4.8.1.1.3.

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D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one D.C. train as listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With a required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required D.C. train to OPERABLE status as soon as possible.

b. With a required charger inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

and, within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to ( ) square inch vent.

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ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160-volt A.C. ESF bus, and three 480-volt A.C. load centers and their associated four class 1E-MCCs.

b. Two 120-volt A.C. channel vital busses energized from their associated inverters connected to their respective D.C. channels.

c. One 125-volt D.C. train with both required channels energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible and within 8 hours depressurize and vent the Reactor Coolant System through a greater than or equal to ( ) square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8-3 shall be bypassed continuously or under accident conditions, as applicable, by an OPERABLE device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously ~~or under accident conditions, as applicable,~~ by an OPERABLE integral bypass device, ~~take administrative action to continuously bypass the thermal overload within 8 hours or~~ declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected ~~system(s)~~. valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 ~~The thermal overload protection for the above required valves which are continuously bypassed shall be verified to be bypassed following testing during which the thermal overload protection was temporarily placed in force.~~

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At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

1. All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

2. All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

and automatically

b. Demonstrated OPERABLE (at least once per 18 months during shutdown by transferring (manually) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

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4. Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 5500 kW in less than or equal to ~~30~~ 60 seconds, and operates with a load greater than or equal to 5500 kW for at least an additional 60 minutes, and X

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks. X

← Insert

d. ~~c.~~ At least once per 92 days and from new fuel prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:

   1. A water and sediment content of less than or equal to 0.05 volume percent;

   2. A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

   3. A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.80 but less than or equal to 0.99 or an API gravity at 60°F of greater than or equal to 11 degrees but less than or equal to 47 degrees;

   4. An impurity level of less than 2 mg of insolubles per 100 mL when tested in accordance with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and

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c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

5. The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.

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e ~~f~~. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of greater than or equal to 885 kW (Train B AFW pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz. (Insert 2)

3. Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed ~~6240~~ 4784 volts during and following the load rejection. X

4. Simulating a loss-of-offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

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d. ~~(For plants with cathodic corrosion protection systems for the ...)~~. Verify that the ~~impressed current type~~ cathodic ~~corrosion~~ protection system is operable at the following time intervals:

1. Verify at least once per 60 days that the cathodic protection rectifiers are operable and have been inspected in accordance with the manufacturer's inspection procedures.
2. Verify at least once per 12 months, that the cathodic protection system is operable and provides adequate protection from corrosion in accordance with manufacturer's inspection procedures.

NOTE: If any other metallic structures (buildings, new or modified piping systems, conduit, etc) are placed in the ground in the vicinity of the fuel oil storage system or if the original system is modified, ~~...~~ the ~~...~~ adequacy and frequency of inspections of the cathodic protection system will be reevaluated and adjusted accordance with the manufacturers recommendations.

Insert 2:

less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above normal which ever is less

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a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential, and low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6050 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 5500 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2~~d~~.6.b. X
.e

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 5500 kW.

9. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

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11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) (turning gear engaged)
    b) (emergency stop)

f ~~e~~. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 600 rpm (steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz) in less than or equal to 10 seconds.

g ~~f~~. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or the equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission within 30 days in a Special Report pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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3/4.8.2 D.C. SOURCES

OPERATNG

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the D.C. trains listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required D.C. trains inoperable, restore the inoperable D.C. trains to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. ~~With one of the required chargers inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(a)	> 2.07 volts
Specific Gravity(b)		$\geq$ 1.195	Not more than 0.020 below the average of all connected cells
	$\geq$ 1.20~~5~~0(c)	Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(c) X

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value, declare the battery inoperable.

(a) Corrected for average electrolyte temperature.

(b) Corrected for electrolyte temperature and level.

(c) Or battery charging current is less than 2 amps when on charge.

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ELECTRICAL POWER SYSTEMS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ESF ~~Load Centers~~ busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. X

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective devices shown in Table 3.8-2 inoperable:

a. Restore the protection device(s) to OPERABLE status or deenergize the circuits(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protection relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-2.

For the lower voltage circuit breakers, the nominal trip setpoint and short circuit response times are listed in Table 3.8-2.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers' nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets it manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8-2

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION	Response Time (sec./cycle) @ trip setpoint
E-NHN-M1006	E-NHN-M1002B	SG WET LAYUP RECIRC. PUMP M-SGN-P01B	
E-NHN-M1017	E-NHN-M1002B	CTMT/RADWASTE SUMP PUMP M-RDN-P03	
E-NHN-M1003	E-NHN-M1002A	RCP 1B CONTROLLED BLEEDOFF VLV J-RCE-HV-431	
E-NHN-M1004	E-NHN-M1002A	RCP 1B HP COOLER INLET VLV J-RCN-HV-447	
E-NHN-M1005	E-NHN-M1002A	RCP 1B HP COOLER OUTLET VLV J-RCN-HV-451	
E-NHN-M1010	E-NHN-M1002A	REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-M02B	
E-NHN-M1014	E-NHN-M1002A	REACTOR CAVITY SUMP PUMP M-RDN-P01A	
E-NHN-M2808	E-NHN-M2832C	RCP 2B CONTROL BLEEDOFF VLV J-RCE-HV-433	
E-NHN-M2813	E-NHN-M2832C	RCP 2B HI PRESSURE COOLER INLET VLV J-RCN-HV-449	
E-NHN-M1009	E-NHN-M1002A	RCP 2B HI PRESSURE COOLER OUTLET VLV J-RCN-HV-453	
E-NHN-M1306	E-NHN-M1314A	SG 2 HOT LEG BLDWN ISO VLV J-SGE-HV-42	
E-NHN-M1307	E-NHN-M1314A	SG 2 COLD LEG BLDWN ISO VLV J-SGE-HV-44	
E-NHN-M1311	E-NHN-M1314D	WET LAY UP RECIRC PUMP M-SGN-P01A	
E-NHN-M1316	E-NHN-M1314C	RCPT (30A) FOR SEAL CRANE ASSY MOTOR E-NHN-122A; E-NHN-122B	
E-NHN-M1339	E-NHN-M1314C	MOVABLE INCORE DETECTOR DRIVE MACHINE M-RIN-M03A	

CF3

ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix, Arizona 85036

received 9/19/84

JND

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

8411030472

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

8410020130

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

Director of Nuclear Reactor Regulation
Page Three

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt / TBK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc
cc: E. Licitra (w/a)
J. B. Martin (w/a)
R. Zimmerman (w/a)
G. Fiorelli (w/a)

PROOF AND REVIEW

CPB

PG 3/4 2-5 CHANGE LCO and Surveillance - 4.2.4.4

JUSTIFICATION: Table 3.3-1 to be submitted later

Action 6 applies to this LCO not just 6C as stated. Also change the LCO to say ...Table 3.3-1. The rest of the sentence can be deleted because it is included in Action 6.

Change 4.2.4.4 to ...once per 31 EFPD because this surveillance refers to a burnup period vs. a calandar period. This change is in compliance with the way PVNGS intends to operate. If the unit is shutdown for any period of time and nothing was done to change the COLSS or CPC DNBR parameters/calculations they do not need to be verified.

PGS 3/4 2-~~8~~ 7 THESE GRAPHS WILL BE SUPPLIED IN ABOUT A WEEK
3/4 2-10

CPB

PGS 3/4 3-7, CHANGE:
3/4 3-8,
3/4 3-7a, See attached.
3/4 3-8b

RSB
CPB

JUSTIFICATION:

Current Arizona Unit 1, Cycle 1 Technical Specifications does not specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEAC's out of service. An analysis was done to specify this lower power level.

This work is the completion of the CEAC's OOS work. This analysis improves ANPP Unit 1, Cycle 1 power capability from about 75% to greater than about 90% with both CEAC's out of service. The analysis of the documents and quality assures this result.

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEAC's out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

PROOF

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature :	580°F
Low Pressure :	1785 psia
ASI :	-.3
Underflow fraction:	0.865
Low Flow :	95% of full flow
High Radial Peak :	1.70 (Bank 5+4+PLR; PDIL = 40% Power; Reference 3)

Conclusion:

This revised Technical Specifications include a minimum power (50% of rated power) below which an additional power reduction is unnecessary, if there is a CEA misalignment with CEAC's out of service.

The minimum power for POL is 70%.

The added statement to action 6a is justified to the justification section for 3/4 1-17.

The work disabled in Items 6.b.1.c and 6.C.1.b are to be replaced with "Placed out of Service". This is consistent with PVNGS Terminology and also there is an indicator on the control panel that states "Out of Service" for the reactor power cutback system.

Delete the terms in 6.b.2.b and 6.C.2.b "...The inoperable status" to read ..."Be indicated that both CEACs are inoperable". This will avoid operator confusion and is consistent with how PVNGS Operators refer to this situation.

PG 3/4 3-9 CHANGE:

See Page.

JUSTIFICATION:

CE number changes.

ICSB
RSB

PG 3/4 3-10 DELETE:

Current # note add new one. See page.

RSB
ICSB

JUSTIFICATION:

The new pound note had been proposed to the NRC for the past 2 years. This change is consistent and reflects the way PVNGS will test the response times. This change is also agreed upon by CE.

"The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input."

The Proof and Review copy, # note reads, "Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown." This requires pertubation of the reactor plant and excludes the capability to utilize test equipment to verify response time. Pertubating the plant is an unnecessary safety and radiological (crud) risk.

PG 3/4 3-10a Type, see page. EPB RSB

PG 3/4 3-12 CHANGE:

III B to read change functional test R. RSB ICSB

JUSTIFICATION:

Page 3/4 3-12, Table 4.3-1 III. B., Channel Functional Test column. Change to read: "R" (delete "M, S/U(1)").

This change is consistent with the SONGS 2 and SONGS 3 Tech Specs.

Table 4.3-1 III. A. provides adequate surveillance requirements for the Reactor Trip Breakers. The difference in plant equipment hardware between Table 4.3-1III. A. and III. B. are the reactor trip pushbutton switches. A surveillance test frequency of "R" is adequate for operability verification of pushbutton switches.

PG 3/4 3-13 ADD

Statement, see page. RSB

JUSTIFICATION:

Add the statement to note 8 for clarification and providing an alternate and more detailed method as to how to determine RCS flow rate.

PG 3/4 3-22 ADD
3-23,
Attachment See page. RSB
3-23-A

JUSTIFICATION:

Add the footnotes as indicated to provide detailed information to the plant staff as what actions to take to avoid confusion. The NR and WR identify if the readings are taken from the Wide Range or Narrow Range instrument.

PROOF AND

power level in the reactor (decrease in coolant temperature and pressure) may aggravate the problem by inducing degassification of the primary coolant.

We are taking fuel building Ventilation exhaust and condenser vacuum plump/gland seal monitor out of Table 4/3-3 because they are duplicated in Table 3.3-13.

PG 3/4 3-39 **CHANGE:** CPB

Surveillance 4.3.3.2b

JUSTIFICATION:

By adding the "7 days or more have elapsed since the last use" provides a time period for guidance for a channel check. This addition is consistent with SONGS tech. specs. They said without this clarification that there were problems with the region and plant interputation.

Delete last sentence of 4.3.3.2.b. We have <u>no</u> means to perform the calibration of the incore detectors. This is a function being performed by all suppliers of incores. Other utilities that have this Tech. Spec. say they cannot really meet this Spec. See justification for Page 3-64-67 on Page 15 for *** justification. –

PG 3/4 3-41 **CHANGE:** SGEB

3.a (setpoint 0.02g).

JUSTIFICATION:

FSAR Section 3.7.4.3, pg. 3.7-32.

PG 3/4 3-44 **CHANGE:** METB

1a and 1b.

JUSTIFICATION:

NUS recently supplied information showed the instrument range to be:

1 to 50 mph
1 to 50 mps

PG 3/4 3-52 **CHANGE:**

The LCO and Applicability. See page. CEB

PROPOSED

PG 3/4 4-32 CHANGE:

LCO and Action Item b. See page.

RSB

JUSTIFICATION:

CE Number Change.

PG 3/4 5-1
5-2

CHANGE:

LCO b. See Page.

RSB
CPB

JUSTIFICATION:

To clarify the Spec. for operations to avoid further confusion. Also the 2000 ppm change is based on a new CE Number.

PG 3/4 5-5 CHANGE:

RSB

(a). See page.

JUSTIFICATION:

CE new number.

PG 3/4 5-6 CHANGE:

RSB

HPSI System - Single Pump. See page.

JUSTIFICATION:

This specification needs to be changed to comply with how we tested the pump during startup as well as how we will test the HPSI pump during operations. This change is in compliance with CE's design criteria.

Change LPSI Pump Sections 1 and 3 from leg to loop.

PG 3/4 6-1 CHANGE:

Surveillance 4.6.1.1a. Table 3.6-0. See pages.

CSB

JUSTIFICATION:

Operations Department requests the attached changes to the PVNGS Technical Specification. Justification for each proposed change is summarized as follows:

Technical Specification 3/4.6.1.1. "Containment Integrity" and Technical Specification 3/4.6.3, "Containment Isolation Valves."

5-6

PG 5-7 5-8 CHANGE:

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

RSB CPB

CH6 CHANGE:

See page.

See next page

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

BASIS CHANGES

See pages.

See next page

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB	B3/4 7-1	RSB ASB
B3/4 1-2	CPB	7-2	RSB ASB
1-3	CPB	7-3	ASB
2-1	CPB	7-4	MEB
3-3	CEB	7-6/7	CEB
3-4	ASB	9-3	AEB ASB
4-1	RSB	10-1	CPB
4-6	RSB	10-2	RSB
4-7	RSB	11-1/2	RAB METB
4-11	RSB	11-5	METB
5-1	CPB		
5-3	RSB		
6-2	CSB		
6-3/4	AEB CSB		

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

DNBR - Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	> 470°F
b.	RCS Cold Leg Temperature-High	< 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	> ~~1868~~ psia 1861
f.	Pressurizer Pressure-High	< ~~2389~~ psia 2388
g.	Integrated Radial Peaking Factor-Low	> 1.28
h.	Integrated Radial Peaking Factor-High	< 4.28
i.	Quality Margin-Low	> 0

Steam Generator Level - High

~~The Steam Generator Level - High trip provides protection in the event of excess feedwater flow. The setpoint for the trip is identical to the main steam isolation setpoint.~~

Reactor Coolant Flow - Low

A FOUR PUMP FLOW COASTDOWN DURING A STEAMLINE BREAK WITH A LOSS OF OFFSITE POWER

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions.

THE STEAM GENERATOR LEVEL-HIGH TRIP IS PROVIDED TO PROTECT THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. SINCE THE TURBINE IS AUTOMATICALLY TRIPPED WHEN THE REACTOR IS TRIPPED, THIS TRIP PROVIDES A RELIABLE MEANS FOR PROVIDING PROTECTION TO THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. THIS TRIP'S SETPOINT DOES NOT CORRESPOND TO A SAFETY LIMIT AND NO CREDIT WAS TAKEN IN THE ACCIDENT ANALYSES FOR OPERATION OF THIS TRIP. ITS FUNCTIONAL CAPABILITY AT THE SPECIFIED TRIP SETTING ENHANCES THE OVERALL RELIABILITY OF THE REACTOR PROTECTION SYSTEM

Pressurizer Pressure - High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications ~~such as~~ of ~~calorimetric measurements for~~ power level, ~~and~~ RCS flow rate, ~~and incore detector signals for~~ axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

PROOF AND REVIEW

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{cold} GREATER THAN 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION: a) WITH KEFF ≥ 1, COMPLY WITH SPECIFICATION 3.1.3.6

b) With the SHUTDOWN MARGIN less than 6.0% delta k/k AND KEFF < 1, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 6.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. IN ADDITION, THE POSITION OF THE PART LENGTH CEAS GROUPS SHALL BE LIMITED TO THE INSERTION LIMITS SHOWN ON FIGURE 3.1-2A

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full-length or part-length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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ACTION: (Continued)

C ~~b~~. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches ~~but less than or equal to 19 inches~~, operation in MODES 1 and 2 may continue, provided that within 1 hour CORE POWER IS REDUCED IN ACCORDANCE WITH FIGURE 3.1-2B AND THAT the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

   a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

D. ~~f~~. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits (Figure 3.1-3) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

~~f~~. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, ensure that the CEA is: (1) within its above specified alignment requirements and, (2) either fully withdrawn or, if in full-length CEA group 5, within the Long Term Steady State Insertion Limits of Figure 3.1-3. Then operation in MODES 1 and 2 may continue.

E ~~g~~. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

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REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, and to the insertion limits## shown on Figure 3.1-3** when the COLSS is in service or shown on Figure 3.1-4** when the COLSS is not in service. The CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Months.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using ~~the above~~ figures 3.1-3 or 3.1-4.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-3 or Figure 3.1-4 are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.

**CEAs are fully withdrawn in accordance with Figure 3.1-3 or Figure 3.1-4 when withdrawn to at least 144.75 inches.

##Following a reactor power cutback in which (1) Regulating Group~~s 4 and/or~~ 5 OR REG. GROUPS AND 5 are dropped or (2) Regulating Groups ~~4 and/or~~ 5 OR REG GROUPS 4 AND 5 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) sequentially inserted, the Transient Insertion Limit of Figure 3.1-3 OR FIGURE 3.1-4 can be exceeded for up to [illegible] hours.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

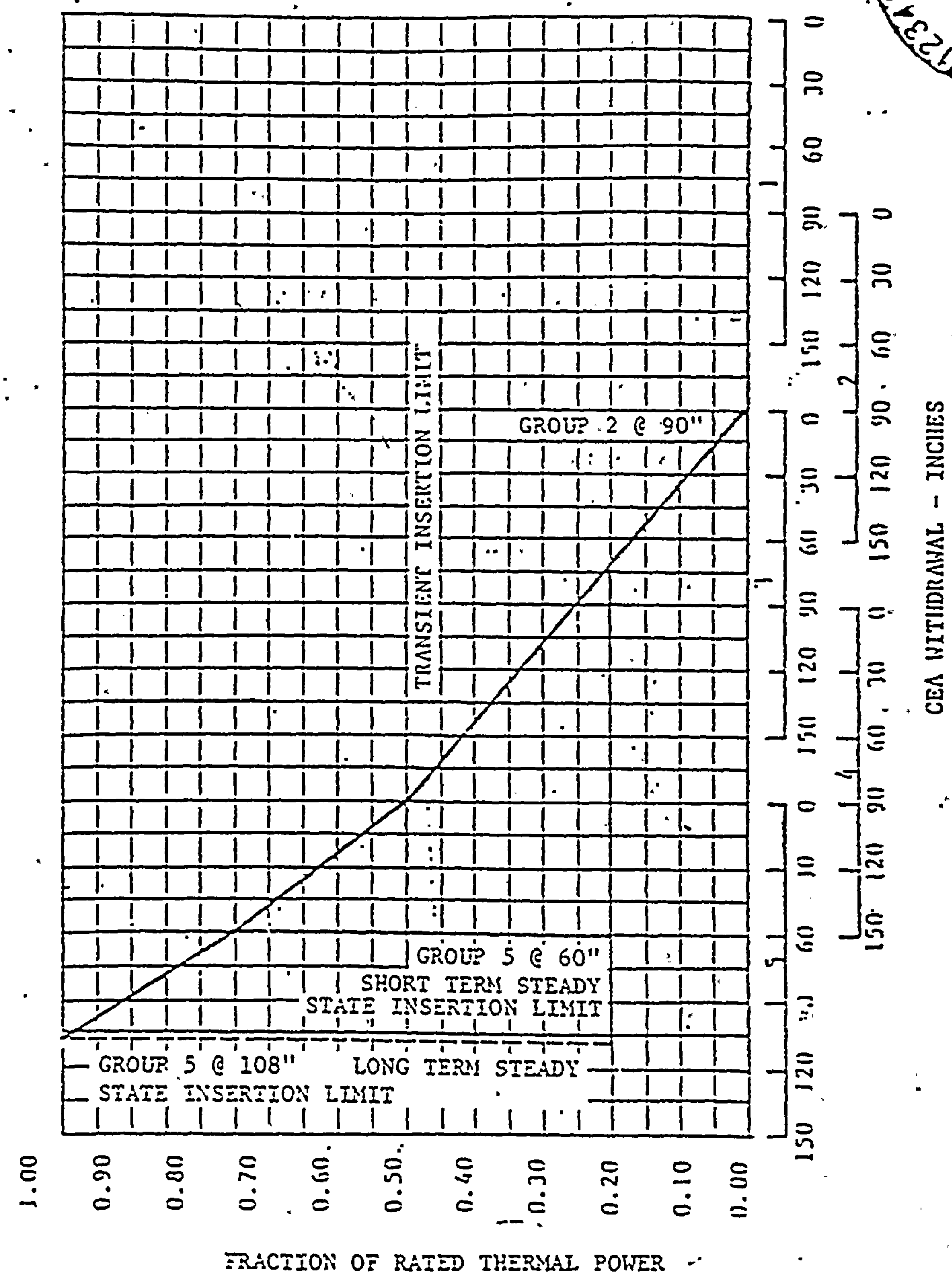
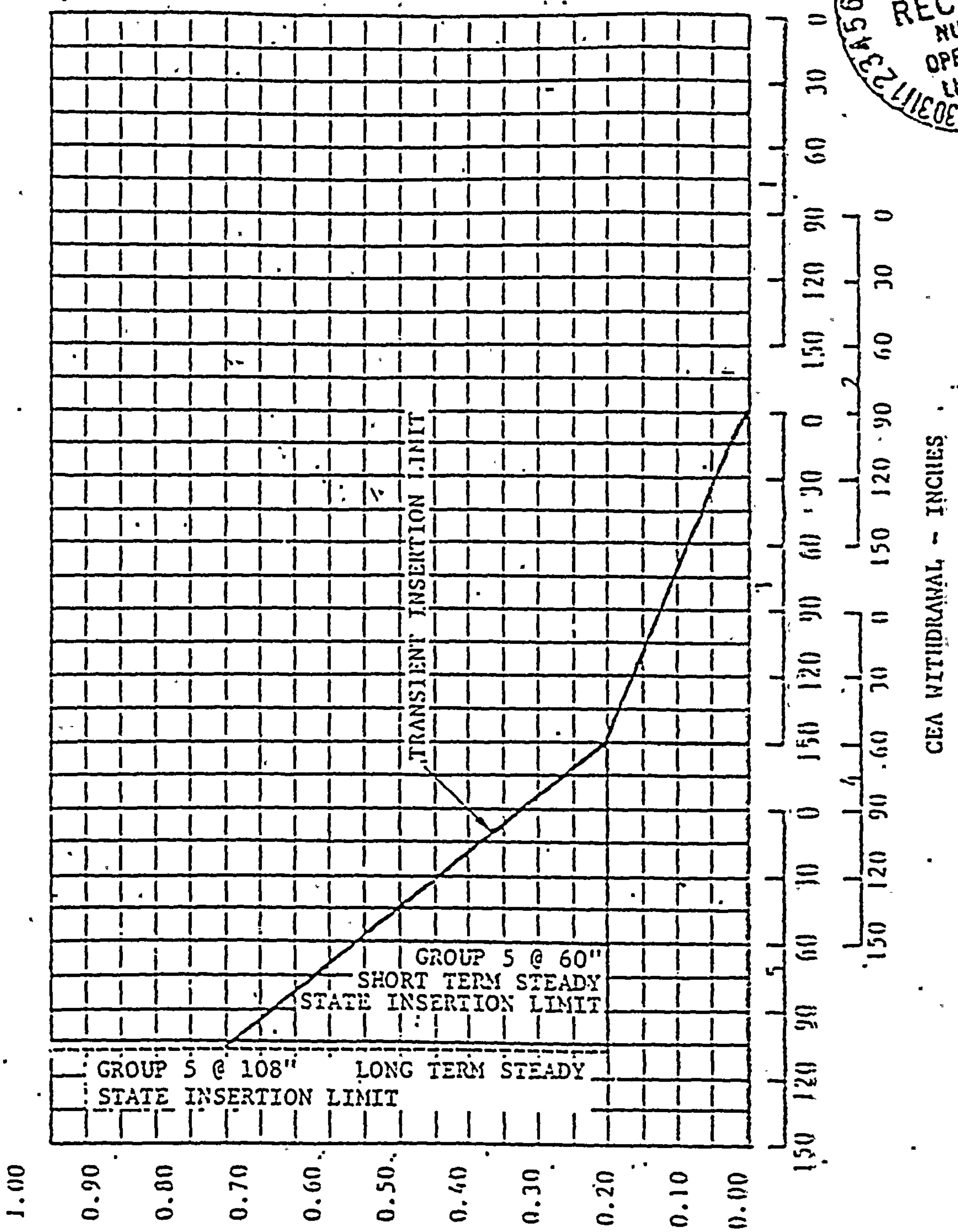


FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS IN SERVICE)



FRACTION OF RATED THERMAL POWER

from V-CE-19028 of 9/22/83

FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS OUT OF SERVICE)

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable, or in accordance with the requirements of Action 6c of Table 3.3-1. ~~when COLSS and both CEACs are inoperable~~.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 ~~days:~~ EFPD.

Burnup $\left(\frac{GWD}{MTU}\right)$	DNBR Penalty (%)*
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

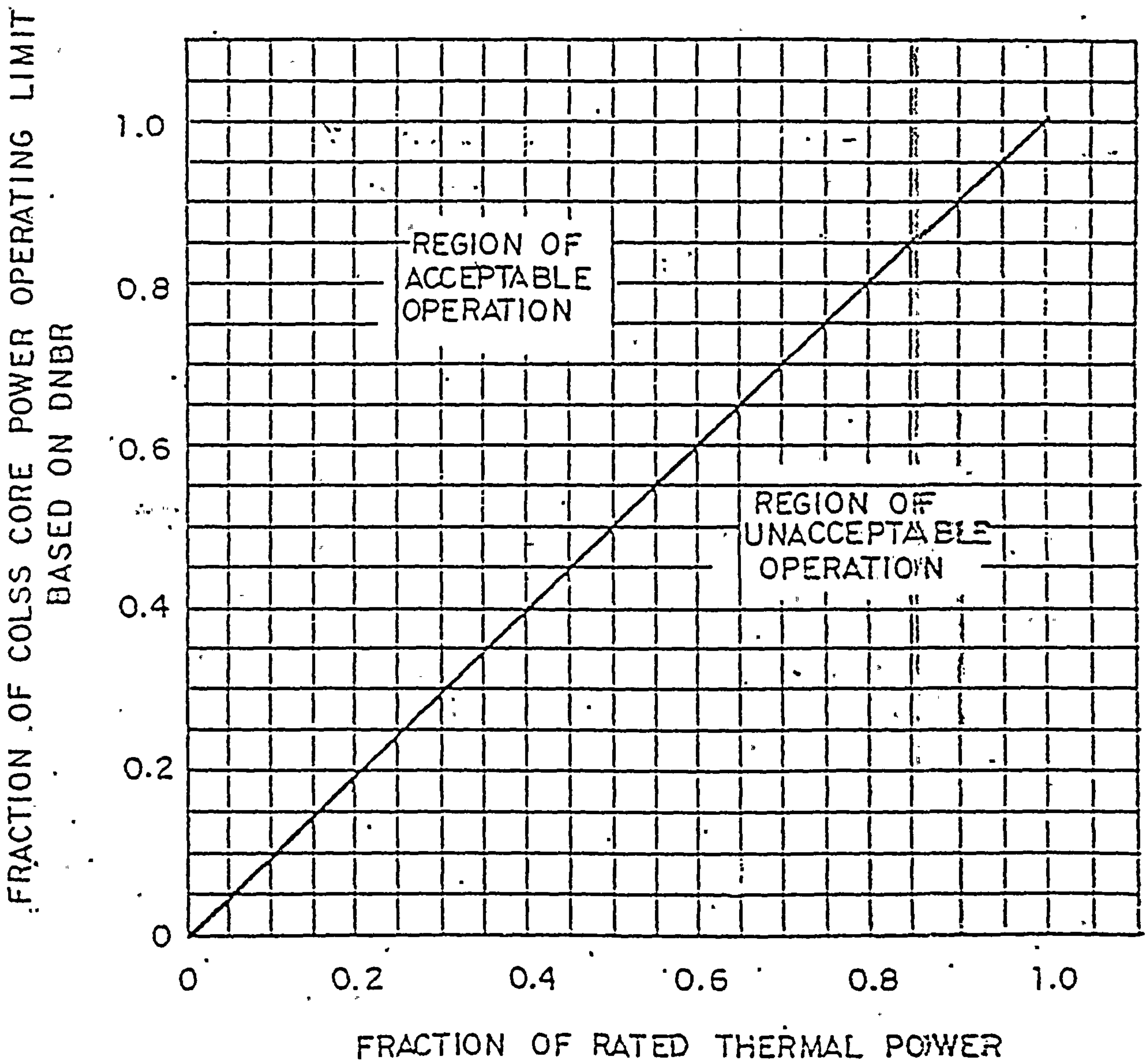


FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS
(COLSS IN SERVICE)

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3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.*

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF).
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1. The trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. AFTER 7 DAYS, OPERATION MAY CONTINUE PROVIDED THAT THE CONDITIONS OF ACTION 6.b OR 6.c ARE MET

b. With both CEACs inoperable and COLSS in ~~operation~~, SERVICE operation may continue provided that:

1. Within 1 hour:

a) The margins required by Specification 3.2.4 are increased and maintained at a value equivalent to or greater than the percentage of RATED THERMAL POWER shown on Figure 3.3-1.

b) The Reactor Power Cutback System is ~~disabled~~. (PLACED OUT OF SERVICE

A) OPERATION IS RESTRICTED TO THE LIMITS SHOWN IN FIG 3.3-1. THE DNBR MARGIN REQUIRED BY SPECIFICATION 3.2.4 IS REPLACED BY THIS RESTRICTION WHEN BOTH CEAC'S ARE INOPERABLE AND COLSS IS IN OPERATION

B) THE LINEAR HEAT RATE MARGIN REQUIRED BY SPECIFICATION 3.2.1 IS MAINTAINED

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status.~~ indicate that both CEACs are inoperable

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA Misalignment, with both CEAC's inoperable and COLSS in operation, operation may continue provided that:
   within 1 hour:
   a) The power is reduced to 80% of the pre-misalignment power but need not be reduced to less than 80% of rated power.
   b) Refer to Section 3.1.3, Movable Control Assemblies, for further specifications on CEA misalignment.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

   1. Within 1 hour:

      a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by ~~1.15~~ 1.19 and the resulting value is re-entered into the CPCs.

      b) The Reactor Power Cutback System is ~~disabled.~~ placed out of service

      c) The COLSS out of service limit line, Section 3.2.4 is not applicable to this mode of operation.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status~~. BE INDICATED THAT BOTH CEAC'S ARE INOPERABLE

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

A. Following a CEA misalignment with both CEAC's and COLSS inoperable, operation may continue provided that:
within 1 hour:
a) The power is reduced to 80% of the pre-misaligned power but need not be reduced to less than 50% of rated power.
b) Refer to Section 3.1.3, Movable Control Assemblies, for further specifications on CEA misalignment

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - Wi . the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour.

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LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

- a. The isolation valve key-locked open and power to the valve removed,
- b: A contained borated water level of between ~~28%~~ (1802 cubic feet) and ~~72%~~ (1914 cubic feet) level (as read on narrow range indication), (i.e. Between 28% and 72%)
- c. A boron concentration between ~~1000~~ 2000 and 4400 ppm of boron, and
- d. A nitrogen cover-pressure of between 600 and 625 psig.
- e. Nitrogen vent valves closed and power removed.**.
- f. Nitrogen vent valves are capable of being operated upon restoration of power.

Not accepted

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APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

- a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

- a. At least once per 12 hours by:
  1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

†With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 60% wide range indication (1415 cubic feet) and 72% narrow range indication (1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 39% wide range indication (962 cubic feet) and 72% narrow range indication (1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.
2000

c. At least once per 31 days when the RCS pressure is above 700 psig, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

   2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 700 psig with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal ~~9.43~~ 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis. With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and A 4% $\Delta k/k$ SHUTDOWN MARGIN requirement~~s are~~ is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

PROOF AND REVIEW

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) an emergency power supply from OPERABLE diesel generators. THE 26 GPM VALUE IS BASED ON NOMINAL FLOW OF ONE CHARGING PUMP LESS THE REACTOR COOLANT PUMP BLEEDOFF FLOW

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

BORATION SYSTEMS (Continued)

The values of water volumes, temperatures, and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analysis remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in MODES 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.2.7 BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at any time, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

With only one startup channel high neutron flux alarm OPERABLE while in MODE 3, 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. ~~This provides a diverse and redundant method of detection of boron dilution with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality~~.

THIS PROVIDES ALTERNATE METHODS OF DETECTION OF BORON DILUTION WITH SUFFICIENT TIME FOR TERMINATION OF THE EVENT BEFORE COMPLETE LOSS OF SHUTDOWN MARGIN AND RETURN TO CRITICALITY

b. Both startup channel high neutron flux alarms not operating:

When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, simultaneous use of boronometer and RCS sampling to monitor the RCS boron concentration provides diverse and redundant indications of an inadvertent boron dilution. ~~This will allow detection with sufficient time for termination of boron dilution before complete loss of SHUTDOWN MARGIN and return to criticality~~.
THEREFORE, EITHER SIMULTANEOUS USE OF THE BORONOMETER AND RCS SAMPLING OR INDEPENDENT COLLECTION AND ANALYSIS OF TWO RCS SAMPLES TO MONITOR

THE RCS BORON CONCENTRATION PROVIDES ALTERNATE INDICATIONS OF AN INADVERTANT BORON DILUTION.

PROOF AND REVIEW

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

PLUS THOSE ASSOCIATED WITH THE CPC STARTUP TEST ACCEPTANCE CRITERIA

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of ~~4000~~ 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

.1 of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1750 psia the Technical Specifications require a minimum of 57% wide range corresponding

PROOF AND REVIEW

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue PHYSICS TESTING.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth, (2) determine the reactor stability index and damping factor under xenon oscillation conditions, (3) determine power distributions for non-normal CEA configurations, (4) measure rod shadowing factors, and (5) measure temperature and power coefficients. Special test exception permits MTC to exceed limits in Specification 3.1.1.3 during performance of PHYSICS TESTS.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

This special test exception permits the CEAs to be positioned beyond the insertion limits and reactor coolant cold leg temperature to be outside limits during PHYSICS TESTS required to determine the isothermal temperatur coefficient and power coefficient.

3/4.10.5 MINIMUM TEMPERATURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{cold} below the minimum critical temperature during PHYSICS TESTS which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions. The Low Power Physics Testing Program at low temperature (300°F) is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. Regulating CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

Core Performance Branch
Comments on Palo Verde 1 T/S memo dated 9/14/85

9/24/84
JND

2. Comments and Suggestions

We have reviewed the above-mentioned sections of the Technical Specifications which pertain to our area of responsibilty and offer the following comments and suggestions:

✓ 1. Safety Limit 2.1.1.1 Comment: See item 5.

✓ 2. Footnote (5) of Table 2.2-1 on page 2-5 should add the following sentences:

The approved SCU-equivalent DNBR limit is 1.231 which includes a rod bow compensation of 0.8 percent DNBR. If fuel burnup exceeds 20,000 MWD/MTU with a rod bow penalty greater than 0.8%, the DNBR limit should be adjusted. A DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{(RB-0.8)}{100} \times \left|\frac{d\,(POL)}{d\,(DNBR)}\right|\right]$$

Where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; POL is the power operating limit; d(POL)/d (DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

✓ 3. The CPC type II addressable constants should be listed in Table 2.2-2.

4. At the end of the third paragraph of Bases 2.1.1 on page B 2-1, add the following sentences:

The DNBR limit of 1.231 includes a rod bow compensation of 0.8 percent on DNBR. For fuel burnups exceeding 20,000 MWD/MTU with a rod bow penalty greater than 0.8 percent DNBR, the DNBR limit should be adjusted.

5. At the end of the second paragraph of Bases 2.2.1, add the following sentences:

The DNBR trip setpoint should be increased for fuel rod bow penalty greater than 0.8 percent DNBR. But the trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.

6. Bases 3.2.5 provides a value for the minimum flow rate required in the Reactor Coolant System but is missing the value for the maximum flow rate permitted.

7. There are two items not listed in Table 3.3-10 for II.F.2, Inadequate Core Cooling Instrumentation:

(a) Reactor Vessel Monitoring System
(b) Core Exit Thermocouples

8. There are two items not listed in Table 4.3-7 for II.F. 2, Inadequate Core Cooling Instrumentation:

(a) Reactor Vessel Monitoring System
(b) Core Exit Thermocouples

9. In Technical Specification 4.1.1.2.2 it is not clear why a k_{eff} of 0.98 is mentioned whereas the required shutdown margin is $\geq 4\% \Delta k$.

10. In Technical Specification 3.1.3.5 the fully withdrawn CEA position is referred to as 144.75 inches or greater. Has the effect of this CEA bite been included in calculating the physics characteristics such as scram worths, shutdown margins, peaking factors, etc.?

11. Why has threshold of above 5% of rated thermal power (STS) been changed to above 20% of rated thermal power in the surveillance requirements of 3.10.2 and 3.10.4?

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER. ← Insert X

(6) RATE is the maximum rate of decrease of the trip setpoint.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

Insert

The approved SCU-equivalent DNBR limit is 1.231 which includes a rod bow compensation of 0.8 percent DNBR. If fuel burnup exceeds 20,000 MWD/MTU with a rod bow penalty greater than 0.8%, the DNBR limit should be adjusted. A DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$\text{BERR1}_{new} = \text{BERR1}_{old} \left[1 + \frac{(\text{RB}-0.8)}{100} \times \left|\frac{d\ (\text{POL})}{d\ (\text{DNBR})}\right|\right]$$

Where BERR1_{old} is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; POL is the power operating limit; d(POL)/d (DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERRO	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
98	CORR1	Temperature shadowing correction factor multiplier
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient

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2.1 and 2.2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to 1.231 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. ← Insert

X

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

Insert

The DNBR limit of 1.231 includes a rod bow compensation of 0.8 percent on DNBR. For fuel burnups exceeding 20,000 MWD/MTU with a rod bow penalty greater than 0.8 percent DNBR, the DNBR limit should be adjusted.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints. ← Insert

X

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Insert

The DNBR trip setpoint should be increased for fuel rod bow penalty greater than 0.8 percent DNBR. But the trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 164.0×10^6 lbm/hr. and less than or equal to ( ) lbm/hr. X

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than its limits at least once per 12 hours. X

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	
1. Containment Pressure	2	1	
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	2	1/loop	
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	2	1/loop	
4. Pressurizer Pressure - Wide Range	2	1	
5. Pressurizer Water Level	2	1	
6. Steam Generator Pressure	2/steam generator	1/steam generator	
7. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	
8. Refueling Water Storage Tank Water Level	2	1	
9. Auxiliary Feedwater Flow Rate	2	1	
10. Reactor Cooling System Subcooling Margin Monitor	2	1	
11. Pressurizer Safety Valve Position Indicator	1/valve	1/valve	
12. Containment Water Level (Narrow Range)	2	1	
13. Containment Water Level (Wide Range)	2	1	
14. Reactor Vessel Level Monitoring System	2	1	X
15. Core Exit Thermocouples	( )/core quadrant	( )/core quadrant	X

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature -T_{cold} (Wide Range)	M	R
4. Pressurizer Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Wide Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Pressurizer Safety Valve Position Indicator	M	R
12. Containment Water Level (Narrow Range)	M	R
13. Containment Water Level (Wide Range)	M	R
14. Reactor Vessel Level Monitoring System	M	R
15. Core Exit Thermocouples	M	R

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SHUTDOWN MARGIN - T_{cold} LESS THAN OR EQUAL TO 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. ~~4.1.1.2.2 K_{eff} shall be determined to be equal to or less than 0.98~~ at least once per 24 hours, when the RCS water level is drained below the pressurizer low level instrument tap, by performing a reactivity balance considering the factors listed in Specification 4.1.1.2.1b.

X

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above ~~20%~~ 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

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SPECIAL TEST EXCEPTIONS

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above ~~20%~~ of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

5%

PROOF AND

PG 3/4 3-64, 65, 66 67.

CHANGES:

Add footnotes.

~~MEEB~~ METB

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability, by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68

DELETE:

ASB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is our belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-~~19~~ 18
4-20

CHANGE:

Tech. Spec. See page.

RSB
ASB
MEB

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 Typo.

PG 3/4 4-29 NEW TABLE WILL BE SUPPLIED IN A WEEK

MEB
RSB
MEB

PG 3/4 4-26

METB

a more detailed action statement to instruct the operators as to what they need to do in situations not identified in the old action statement.

PG 3/4 6-37 CHANGE: ME

Surveillance 4.6.4.2.b.1. Delete all.

JUSTIFICATION:

The word "ALL" needs to be deleted. There is some instrumentation that is not "vital" or needed to insure proper operation of the recombiner to perform its intended safety functions.

PG 3/4 6-38 CHANGE:

LCO Delete the last six words of the LCO ... "In each of the three units." METB

JUSTIFICATION:

This statement is confusing in its original form. Our people interpret it to mean that power needs to be supplied from all three units no matter where the containment hydrogen purge cleanup system is located. The proposed change clarifies this problem.

PG 3/4 6-39 ADD: METB

Page. This page was deleted in the Proof/Revision copy of the Tech. Spec.

PG 3/4 7-2,3 CHANGE: ASB

See page.

JUSTIFICATION

CE supplied input.

PG 3/4 7-4
7-5 CHANGE: ASB

Surveillance Requirements 4.7.1.2.a.1, 4.7.1.2.a.3, and 4.7.1.2.c. See pages.

JUSTIFICATION:

Delete the last sentence in 4.7.1.2.1.1. This statement will be broken out into item 4.7.1.2.d. This is done to add clarification.

PG3/4 7-1 me

PG3/4 7-9a "Atmosphere Dump Valves New T/S 3.7.1.6

RSB
ASB

ADD:

..., "sealed, or otherwise secured in at the open position." This statement was added to comply with the way PVNGS does business.

ASB

PG 3/4 7-6 CHANGE:

Applicability. See page.

JUSTIFICATION:

The footnote was added to identify that when cooldown is in progress that the LCO in modes 3 and 4 do not require that the condensate storage tank does not require 300,000 gallons of water. This has been verified with our Safety Analysis.

PG 3/4 7-9 CHANGE:

ASB

See page.

JUSTIFICATION

This input was supplied by the valve manufacturer.

PG 3/4 7-10 CHANGE:

ASB

Surveillance. -

Change once per hour to once per shift.

JUSTIFICATION:

During our startup testing this spec required someone to take hourly data. It was noticed that there was not much change in temperature and pressure conditions when taken hourly and compared to an 8 hour shift.

PG 3/7 7-16 CHANGE:

~~ASB~~
~~ASB~~
AEB

Surveillance Requirement 4.7.7.d.3.
... equal 1/8 inch ...

JUSTIFICATION:

This is the value Bechtel recommends. We did meet this during startup testing. This is also in compliance with the Reg Guide.

7-14 See page 33

ASB
AEB

PG 3/4 8-22 Typo's. PSB

PG 3/4 8-23/24/25 Typo's. PSB

PG 3/4 8-26 Typo's. PSB

PG 3/4 8-27 CHANGE: PSB

Table 3.8-2. Delete control panel CEDM M-G Set J-SFN-C02B.

JUSTIFICATION:

This can be deleted based on the fact that the individual CEAs penetrations are Tech Speced in Table 3.8-2. One of the other M-G sets was taken out in the pre proof/review copy of the Tech Specs. This really appears to be a typo.

PG 3/4 9-4 CHANGE: CSB

Surveillance 4.9.4.b.
... "Portions of Specification 4.9.9."

JUSTIFICATION:

The way the present Tech. Spec. is written it allows you to test the containment purge valves per applicable section of 4.6.3.2. This allows a potential for error in that one may not interpret the "Applicable" part of 4.6.3.2 the same as another. We believe that by specifying Spec 4.9.9 there is no allowance for error as in interpreting the applicable portion of a Tech. Spec. Spec 4.9.9 spells out what must be tested and the period for testing.

PG 3/4 9-6 CHANGE: ASB

See page.

JUSTIFICATION:

This Spec has been revised to use the correct terminology and limits associated with the PVNGS refueling machine.

CEA deletion is justified in that the major concern in this Tech Spec is damage to a fuel assembly. CEAs are not a problem compared to the damaged fuel assembly problem.

3/4 8-27a,b,c,d,e,f justification see Page 35 PSB

PG 3/4 9-12 **DELETE:** me

See page.

JUSTIFICATION:

Typo.

PG 3/4 9-13 **CHANGE:** ASB AEB

LCO. See page.

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 **CHANGE:** RSB

LCO Item C. See page. me

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 **CHANGE:** RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 **CHANGE:** ~~[illegible]~~

Table 4.11-1. See page. METB

JUSTIFICATION:

Number left off Table.

PG 3/4 11-3 **CHANGE:** METB

Footnote b. See page.

discharge to the environment. Both of these discharge pathways pass through the plant vent. An increase in the noble gas monitor count rate of greater than a factor of 3 would indicate that the isotopic mixture of the effluent had change sufficiently to warrant a re-analysis of alarm setpoints and projected offsite dose. Grab samples of noble gas and Tritium are analyzed weekly for the generation of Gaseous Effluent Release Permits. It is felt that the weekly sampling frequency in addition to the Containment Atmosphere Monitor readings will monitor plant vent discharges so they will remain below Technical Specification and 10CFR20 limits. The Fuel Building Exhaust discharge should not be affected by power level changes.

METB

PG 3/4 11-15 CHANGE:

Surveillance 4.11.2.6. See page.

METB
RAB

JUSTIFICATION:

Add ... "During Addition" to the Surveillance Requirement. This is needed so it identifies how we will perform this surveillance more accurately thus, avoiding possible errors. Since this spec is for quantity of radioactivity in the gas storage tank, it should only be monitored during additions to that tank.

PG 3/4 11-17 CHANGE:

LCO. See page.

METB

JUSTIFICATION:

PVNGS does not have filter sludges; therefore, this reference in the LCO is not applicable to the PVNGS design.

PG 3/4 11-18 CHANGE:

Surveillance 4.11.4.1. See page.

RAB

JUSTIFICATION:

Delete all references to liquid effluent for the justification stated above for 3.11.1.2.

PG 3/4 5-6 CHANGE:

5.6.1.1.b.

ASB

~~B3/4 1-1~~ ~~CPB~~

PROOF AND REVIEW

JUSTIFICATION:

Number change to 9.5 to agree with FSAR.

PG 3/4 7-14 CHANGE:

Action Statement. See page.

ASB

JUSTIFICATION

The Essential Chilled Water System, in conjunction with the respective emergency HVAC units, is required (according to Technical Specification definition 1.18) to provide heat removal in maintaining the various Engineered Safety Feature (ESF) room space design temperatures below the associated equipment qualification limits for the range of design basis accident conditions. The normal HVAC system is redundant to the emergency HVAC system in maintaining the space design conditions of required safety systems during normal operating conditions and design basis accident conditions not involving seismic events or loss of offsite power. A 7 day action requirement is proposed for a single ECWS out of service, based on the high reliability of offsite power and availability of the normal HVAC system (the normal HVAC system contains two 100 percent redundant chillers, with one chiller normally in continuous operation and the other in standby). Action requirements are provided to ensure OPERABILITY of the vital bus inverters and emergency battery chargers, by verifying within one hour that the normal HVAC system is providing space cooling to the vital power distribution rooms (loss of space cooling will not result in loss of vital bus inverter or emergency battery charger function in less than 75 minutes). Probablistic risk assessment studies at similar plants have shown that seismic events and events involving fire are the major contributors to accident risk. Therefore, an ACTION requirement is provided to establish within 8 hours operability of the safe shutdown systems which do not depend on the inoperable ECWS. The 8 hour period provides a reasonable time in which to establish OPERABILITY of this complement of key safety systems. This requirement ensures that a functional train of safe shutdown equipment is available to put the plant in a safe, stable condition for the most probable abnormal operational occurrences. An action requirement of 24 hours is provided to establish operability of the remaining required safety systems which do not depend on the inoperable ECWS. This period permits completion of maintenance activities and/or activation of third-of-a-kind components to ensure that an OPERABLE train of safety systems is available to mitigate the range of design basis events during the remainder of the Action.

PG 5-7 5-8	CHANGE:	RSB. CPB

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

CH6	CHANGE:	See next page

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

BASIS	CHANGES	See next page

See pages.

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

Bases

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB	B3/4 7-1	RSB ASB
B3/4 1-2	CPB	7-2	RSB ASB
1-3	CPB	7-3	ASB
2-1	CPB	7-4	MEB
3-2	METB	7-6/7	CEB
3-3	CEB	9-3	AEB ASB
3-4	ASB	10-1	CPB
4-1	RSB	10-2	RSB
4-6	RSB MtEB	11-1/2	RAB METB
4-7	RSB MtEB	11-5	METB
4-11	RSB MtEB		
5-1	CPB		
5-3	RSB		
6-2	CSB		
6-3/4	AEB CSB		

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

   1. Four high pressure turbine stop valves.
   2. Four high pressure turbine control valves.
   3. Six low pressure turbine reheat stop valves.
   4. Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of ~~2225~~ 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve or within 72 hours following a system response to an Engineered Safety Feature actuation signal.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

*TESTING PER SPECIFICATION 4.4.5.2.2.d IS NOT APPLICABLE DUE TO POSITIVE INDICATION OF VALVE POSITION IN THE CONTROL ROOM

#1. LEAKAGE RATES LESS THAN OR EQUAL TO 1.0 GPM ARE CONSIDERED ACCEPTABLE

#2. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED ACCEPTABLE IF THE LATEST MEASURED RATE HAS NOT EXCEEDED THE RATE DETERMINED BY PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

#3. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED UNACCEPTABLE IF THE LATEST MEASURED RATE EXCEEDED THE RATE DETERMINED BY THE PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

#4. LEAKAGE RATES GREATER THAN 5.0 GPM ARE CONSIDERED UNACCEPTABLE.

PROOF AND REVIEW

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE	DESCRIPTION
1) SIV 237	LOOP 1A RC/SI CHECK
2) SIV 247	LOOP 1B RC/SI CHECK
3) SIV 217	LOOP 2A RC/SI CHECK
4) SIV 227	LOOP 2B RC/SI CHECK
5) SIV 235	LOOP 1A SIT CHECK
6) SIV 245	LOOP 1B SIT CHECK
7) SIV 215	LOOP 2A SIT CHECK
8) SIV 225	LOOP 2B SIT CHECK
9) SIV 542	LOOP 1A SI HEADER CHECK
10) SIV 543	LOOP 1B SI HEADER CHECK
11) SIV 540	LOOP 2A SI HEADER CHECK
12) SIV 541	LOOP 2B SI HEADER CHECK
13) SIV 522	LOOP 1 HP LONG TERM RECIRCULATION CHECK
14) SIV 523	LOOP 1 HP LONG TERM RECIRCULATION CHECK
15) SIV 532	LOOP 2 HP LONG TERM RECIRCULATION CHECK
16) SIV 533	LOOP 2 HP LONG TERM RECIRCULATION CHECK
17) UV 651 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
18) UV 652 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION
19) UV 653 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
20) UV 654 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a: With both reactor coolant loops and associated steam generators in operation and with one or more** main steam line code safety valves inoperable per steam generator, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam line code safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, heaters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the fuel building at a slight negative pressure relative to the outside atmosphere during system operation.

(measurable negative pressure)

PROOF AND REVIEW

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the ~~maximum~~ allowable fracture toughness stress limits. The limitations to ~~(90)~~°F and 120°F 230 ~~(275)~~ psig are based on a steam generator RT_{NDT} of ~~(30)~~°F 40 and are sufficient to prevent brittle fracture.

3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the essential cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

The OPERABILITY of the essential spray pond system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 27-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the essential chilled water system ensures that sufficient cooling capacity is available for continued operation of equipment and control room habitability during accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

The OPERABILITY of the control room essential filtration system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (23 feet) is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

A nominal

3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

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3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate ~~radioactivity~~ monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere (gaseous and particulate) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Within 24 hours ~~Prior to entering MODE 2~~ following valve actuation due to automatic or manual action or flow through the valve ~~or~~ Within 72 hours following a system response to an Engineered Safety Feature actuation signal.

e.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam line code safety valves inoperable per steam generator operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam line code safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following ~~any cold shutdown~~ Mode 5 or 6 of 30 days or longer, by verifying (by means of a flow test) the normal flow path from the condensate storage tank to each of the steam generators through each of the auxiliary feedwater pumps. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 23 feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the auxiliary feedwater pumps by verifying:

a. That the as reactor makeup water tank supply line to the ~~to~~ auxiliary feed system isolation valves ~~are~~ open, and

b. That the reactor makeup water tank contains a water level~~s~~ of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least ~~100~~ 144 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~100~~ 144 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least ~~22~~ 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the nitrogen accumulator at a pressure ≥ 400 psig is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV which holds the plant at hot standby while dissipating core decay heat or which allows a flow of sufficient steam to maintain a controlled reactor cooldown rate. A pressure of 400 psig retains sufficient nitrogen volume for 4 hours of operation at hot standby plus 6.5 hours of operation to reach cold shutdown under natural circulation conditions in the event of failure of the normal control air system.

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These values are not in agreement with the FSAR which states 875 gpm @ 3300 [illegible]

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 987 gpm at a pressure of 1260 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 987 gpm at a pressure of 1260 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY for 4 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with ~~concurrent~~ total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to (90)°F and (275) psig are based on a steam generator RT_{NDT} of (30)°F and are sufficient to prevent brittle fracture.

3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the essential cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

The OPERABILITY of the essential spray pond system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 27-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974. intent of the

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the essential chilled water system ensures that sufficient cooling capacity is available for continued operation of equipment and control room habitability during accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

The OPERABILITY of the control room essential filtration system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

±

VALVE NUMBER S/G No. 1	S/G No. 2	LIFT SETTING (±1%)*	MINIMUM RATED CAPACITY**
a. PSV 572	PSV 554	1250 psig	941,543 lb/hr
b. PSV 579	PSV 561	1250 psig	941,543 lb/hr
c. PSV 573	PSV 555	1290 psig	971,332 lb/hr
d. PSV 578	PSV 560	1290 psig	971,332 lb/hr
e. PSV 574	PSV 556	1315 psig	989,950 lb/hr
f. PSV 575	PSV 557	1315 psig	989,950 lb/hr
g. PSV 576	PSV 558	1315 psig	989,950 lb/hr
h. PSV 577	PSV 559	1315 psig	989,950 lb/hr
i. PSV 691	PSV 694	1315 psig	989,950 lb/hr
j. PSV 692	PSV 695	1315 psig	989,950 lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation. These capacities provide a minim[illegible] total capacity of 19,530,900 lb/hr at 1355 psig (1315 psig + 3% accumulation).

STEADY STATE POWER LEVEL AND TABLE 3.7-2

MAXIMUM ALLOWABLE VARIABLE OVERPOWER HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING TWO LOOP OPERATION WITH FOUR PUMPS OPERATING

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)	MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (PERCENT OF RATED THERMAL POWER)
1	108.0	98.2
2	97.1	87.3
3	86.2	76.4
4	75.3	65.5

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each AFW pump, and the manual discharge line valve of each AFW pump are locked, ~~in the open position~~ SEALED, OR OTHERWISE SECURED IN THE OPEN POSITION.

*Until the steam generators are no longer required for heat removal.

SURVEILLANCE REQUIREMENTS (Continued)

b.. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

CAPS refueling shutdown or that

c. Prior to startup following any cold shutdown of 30 days or longer, by verifying (by means of a flow test) the normal flow path from the condensate storage tank to each of the steam generators through each of the auxiliary feedwater pumps. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump. or MODE 4

d.

delivers 750 gpm at 1250 psig or equivalent

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 23 feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3#, and 4*#.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank to auxiliary feed system isolation valves are open, and

b. That the reactor makeup water tank contains water levels of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

NOT APPLICABLE WHEN COOLDOWN IS IN PROGRESS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

IN MODE 2

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least ~~HOT STANDBY~~ within the next 6 hours; ~~and in COLD SHUTDOWN within the following 30 hours~~.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 4 to perform the surveillance testing of Specification 4.7.1.5.1 provided the testing is performed within 12 hours after achieving NORMAL ~~sufficient~~ steam pressure to perform the test.
OPERATING AND NORMAL OPERATING TEMPERATURE FOR THE SECONDARY SIDE
MODE 3 OR

NEW

PLANT SYSTEMS

ATMOSPHERE DUMP VALVES

LIMITING CONDITIONS FOR OPERATIONS

3.7.1.6 The atmospheric dump valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*

ACTION:

With less than one atmospheric dump valve per steam generator OPERABLE, restore the required atmospheric dump valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying nitrogen accumulator tank at a pressure ≥ 400 PSIG.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer verify that all valves will open and close fully.

* When steam generators are being used for decay heat removal.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 120°F when the pressure of the secondary coolant in the steam generator is greater than 230 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 230 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in the secondary side of the steam generators shall be determined to be less than 230 psig at least once per hour when the temperature of the secondary coolant is less than 120°F.

SHIFT

PROOF AND REVIEW

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 At least two independent essential chilled water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

7 DAYS

a) With only one essential chilled water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6 At least two essential chilled water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b) WITH ONLY ONE EMERGENCY CHILLED WATER SYSTEM OPERABLE:

1. WITHIN 1 HOUR VERIFY THAT THE NORMAL HVAC SYSTEM IS PROVIDING SPACE COOLING TO THE VITAL POWER DISTRIBUTION ROOMS THAT DEPEND ON THE INOPERABLE ESSENTIAL CHILLED WATER SYSTEM FOR SPACE COOLING, AND
2. WITHIN 8 HOURS ESTABLISH OPERABILITY OF THE SAFE SHUTDOWN SYSTEMS WHICH DO NOT DEPEND ON THE INOPERABLE ESSENTIAL CHILLED WATER SYSTEM (ONE TRAIN EACH OF BORATION, PRESSURIZER HEATERS AND AUXILIARY FEEDWATER)
3. WITHIN 24 HOURS ESTABLISH OPERABILITY OF ALL REQUIRED SYSTEMS, SUBSYSTEMS, TRAINS, COMPONENTS AND DEVICES THAT DEPEND ON THE REMAINING OPERABILITY ESSENTIAL CHILLED WATER SYSTEM FOR SPACE COOLING

IF THESE CONDITIONS ARE NOT SATISFIED WITHIN THE SPECIFIED TIME, BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS IN COLD SHUTDOWN WITHIN THE FOLLOWING 30 HOURS.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

REFUELING OPERATIONS

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3590 (3990)* pounds and an overload cut off limit of less than or equal to 1556 ~~(1736)~~* pounds for the fuel mast. Refueling machine 1727.

b. A minimum capacity of 2000 pounds and an overload cut off limit of less than or equal to 1651 (1831)* pounds for the CEA mast.

APPLICABILITY: During movement of CEAs or fuel assemblies within the ~~reactor pressure vessel~~ Refueling cavity

ACTION:

a. With the above requirements for the fuel mast Refueling machine not satisfied, suspend use of the fuel mast Refueling machine from operations involving the movement of fuel assemblies.

b. With the above requirements for the CEA mast not satisfied, suspend use of the CEA mast from operations involving the movement of CEAs.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The fuel mast Refueling machine used for movement of fuel assemblies shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3590 (3990)* pounds and demonstrating an automatic load cut off when the fuel mast Refueling machine load exceeds 1556 ~~(1736)~~* 1727 pounds.

4.9.6.2 The CEA mast used for movement of CEAs shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 2000 pounds and demonstrating an automatic load cut off when the CEA mast load exceeds 1651 (1831)* pounds.

*For initial fuel load only.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least ~~23~~ 22'-8" feet of water shall be maintained over the top of ~~irradiated fuel assemblies seated in~~ the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal ~~9.43~~ 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

PROOF AND REVIEW

DELETE REPLACE WITH ATTACHED PAGE

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1381 psig) of its design pressure of 1256 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 19.53×10^6 lb/hr which is 113% of the total secondary steam flow of 17.18×10^6 lb/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two-loop, or four-pump operation

$$SP = \left(\frac{10-N}{10}\right) \times 113$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = total number of secondary safety valves for one steam generator.

N = the number of inoperable secondary safety valves on the steam generator with the greater number of inoperable valves.

113 = the ratio of the total relieving capacity of all twenty (20) secondary safety valves (19.53×10^6 lb/hr at 1355 psig, maximum set pressure plus 3%, accumulation) over the secondary steam flow at 100% Rated Thermal Load (17,180,000 lb/hr).

The main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes, a turbine trip (without reactor trip or cutback) from RATED THERMAL POWER with a coincident loss of condenser heat sink (i.e., no steam bypass) is assumed. The combined relieving capacity of the pressurizer safety valves, and the heat removal capacity of the MSSVs, is sufficient to maintain the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks and 110% of design pressure for all other events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity of all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity of 19.53×10^6 lbm/hr given in Table 3.7-1 as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainity) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity of 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

Although the variable high power reactor trip is not relied on for the limiting overpressure events, the ceiling on this trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

SP = Allowable Power Level + 9.8

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is the ratio of the available relieving capacity of the total steam flow at rated power

10 = total number of main steam safety valves for one steam generator

N = number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves

109.2 = ratio of main steam safety valve relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainity (see above text)

9.8 = BAND between the maximum thermal power and the variable overpower trip setpoint ceiling

PROOF AND REVIEW

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ~~with the~~ ENSURES THAT A minimum WATER volume OF 300,000 GALLONS ~~ensures that sufficient water~~ is available to maintain the Reactor Coolant System at HOT STANDBY for ~~24~~ 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

PG 3/4 6-12, 13 and 14 DELETE:

See page.

JUSTIFICATION:

These pages applied to the Old Tondon Tech. Spec. not to the Tech. Spec. in the proof/review copy. It should have been deleted last revision. The deleted information in the old tables shows up in our new table.

PG 3/4 6-15 CHANGE:

AEB

Action C. See page.

JUSTIFICATION:

Add the following ... OPERABLE status "or isolate the penetrations" within 24 hours ...

This addition is to bring Action C into compliance with Actions A and B.

PG 3/4 6-16 Typo's. Surveillance Req. 4.6.2.1.C.

PG 3/4 6-20, 6-21 CHANGE: LCO - Add * Asterisk.

LCO - Add * Asterisk. See page.

JUSTIFICATION:

The statement was added to clarify to the operators exactly what needs to be done. This change was prompted by a difference in interpretation between our region and an operating utility.

PG 3/4 6-22, 6-35 CHANGE:

Table 3.6-1. See page.

JUSTIFICATION:

See Justification provided in 3.6-1.

PG 3/4 6-36 CHANGE:

Action Statements. See page.

JUSTIFICATION:

The action of this Tech. Spec. based on problems at SONGS. These proposed and approved Tech. Spec. provides

a more detailed action statement to instruct the operators as to what they need to do in situations not identified in the old action statement.

PG 3/4 6-37 CHANGE: CSB

Surveillance 4.6.4.2.b.1. Delete all.

JUSTIFICATION:

The word "ALL" needs to be deleted. There is some instrumentation that is not "vital" or needed to insure proper operation of the recombiner to perform its intended safety functions.

PG 3/4 6-38 CHANGE: AEB

LCO. Delete the last six words of the LCO ... "In each of the three units."

JUSTIFICATION:

This statement is confusing in its original form. Our people interpret it to mean that power needs to be supplied from all three units no matter where the containment hydrogen purge cleanup system is located. The proposed change clarifies this problem. AEB METB

PG 3/4 6-39 ADD:

Page. This page was deleted in the Proof/Revision copy of the Tech. Spec.

PG 3/4 7-1 7-2,3 CHANGE: ASB

See page.

JUSTIFICATION

CE supplied input.

PG 3/4 7-4 7-5 CHANGE: ASB

Surveillance Requirements 4.7.1.2.a.1, 4.7.1.2.a.3, and 4.7.1.2.c. See pages.

JUSTIFICATION:

Delete the last sentence in 4.7.1.2.1.1. This statement will be broken out into item 4.7.1.2.d. This is done to add clarification.

ADD:

..., "sealed, or otherwise secured in at the open position." This statement was added to comply with the way PVNGS does business.

PG 3/4 7-6 CHANGE: ASB

Applicability. See page.

JUSTIFICATION:

The footnote was added to identify that when cooldown is in progress that the LCO in modes 3 and 4 do not require that the condensate storage tank does not require 300,000 gallons of water. This has been verified with our Safety Analysis.

PG 3/4 7-9 CHANGE: ASB

See page.

JUSTIFICATION

This input was supplied by the valve manufacturer.

PG 3/4 7-10 CHANGE: ASB

Surveillance.

Change once per hour to once per shift.

JUSTIFICATION:

During our startup testing this spec required someone to take hourly data. It was noticed that there was not much change in temperature and pressure conditions when taken hourly and compared to an 8 hour shift.

PG 3/7 7-16 CHANGE:

Surveillance Requirement 4.7.7.d.3. AE
... equal 1/8 inch ...

JUSTIFICATION:

This is the value Bechtel recommends. We did meet this during startup testing. This is also in compliance with the Reg Guide.

7-1A See page 33 ASB

PG 3/4 9-12 DELETE: me

See page.

JUSTIFICATION:

Typo.

PG 3/4 9-13 CHANGE: ASB AEB

LCO. See page.

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 CHANGE: RSB me

LCO Item C. See page.

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 CHANGE: RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 CHANGE: ~~AEB~~ METB

Table 4.11-1. See page.

JUSTIFICATION:

Number left off Table.

PG 3/4 11-3 CHANGE: METB

Footnote b. See page.

PG 5-7 5-8 CHANGE:

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

CH6 CHANGE:

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

BASIS CHANGES

See pages.

JUSTIFICATION

The basis has been revised in parts to more accurately reflect our plant.

RSB
CPB

AEB
6.24

See next page

See next page

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB	B3/4 7-1	RSB ASB
B3/4 1-2	CPB	7-2	RSB ASB
1-3	CPB	7-3	ASB
2-1	CPB	7-4	MEB
3-3	CEB	7-6/7	CEB
3-4	ASB	9-3	AEB ASB
4-1	RSB	10-1	CPB
4-6	RSB	10-2	RSB
4-7	RSB	11-1/2	RSB METB
4-11	RSB	11-5	METB
5-1	CPB		
5-3	RSB		
6-2	CSB		
6-3/4	AEB CSB		

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 42-inch containment purge supply and exhaust isolation valve shall be sealed closed.
b. The 8-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 3000 hours per 365 days.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for more than 3000 hours per 365 days, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specifications 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

OR ISOLATE THE PENETRATIONS

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 42-inch containment purge supply and exhaust isolation valves shall be verified to be sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 8-inch purge supply or exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 42-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.4 At least once per 92 days each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.01 L_a when pressurized to P_a.

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus. ~~in each of the three units~~.

APPLICABILITY: MODES 1* and 2.*

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

*With less than two hydrogen recombiners OPERABLE.

3. Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

8.4

PRE FILTERS 1. Verifying that the pressure drop across the combined HEPA filters, and charcoal adsorber banks is less than ~~(2)~~ inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%.

AT LEAST

2. Verifying that the heaters dissipate 0.5 ~~± 0.1~~ kW when tested in accordance with ANSI N510-~~1975~~ 1980

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%.

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room essential filtration systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room essential filtration system in the recirculation mode.

b. With both control room essential filtration systems inoperable, or with the OPERABLE control room essential filtration system, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room essential filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

PROOF AND REVIEW

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a control room essential filtration actuation, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation.

1/8

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least ~~23~~ 22'-8" feet of water shall be maintained over the top of ~~irradiated fuel assemblies seated in~~ the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

PROCEDURES AND PROGRAMS (Continued)

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each program or procedure of specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5.1 and approved prior to implementation. Programs and administrative control procedures shall be approved by the Director of Nuclear Operations, or designated alternate. Implementing procedures shall be approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom is the Shift Supervisor or Assistant Shift Supervisor on the unit affected.

c. The change is documented, reviewed in accordance with Specification 6.5.1 and approved by the Director of Nuclear Operations or congnizant department head, as designated by the Director of Nuclear Operations, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the congnizance of the NSG at least once per 24 months:

a. PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

A program to reduce leakage from those portions of systems outside ocntainment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels: The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

PROCEDURES AND PROGRAMS Continued)

b. In-Plant Radiation Monitoring

A program will be written and issued prior to system being declared operational which will ensure the capability to accurately determine the airborne iodine concnetration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment. Operational

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required ot initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

PROCEDURES AND PROGRAMS (Continued)

e. Post-accident Sampling

A Program will be written and issued prior to system being declared operational which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear; thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

PROOF AND REVIEW

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure ~~a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes~~ ADEQUATE CHEMICAL AVAILABLE TO REMOVE IODINE FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA.

~~IODINE REMOVAL SYSTEM (Continued)~~

~~the evolution of iodine and minimizes the effect of chloride stress corrosion and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety~~ analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (23 feet) is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

A nominal

3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

Accident Evaluation Branch:
comments on Palo Verde 1 T/S
dated 9/10/84

Include T/S 4.7.7.a to limit maximum temperature in control room to protect at least equipment.

Include makeup flow rate into control room in T/S 4.7.7.d.3 where license verifys the positive pressure of control room. The Standard Review Plan requires the positive pressure is with respect to adjacent areas <u>not</u> the outside atmosphere.

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3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room essential filtration systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room essential filtration system in the recirculation mode.

b. With both control room essential filtration systems inoperable, or with the OPERABLE control room essential filtration system, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room essential filtration system shall be demonstrated OPERABLE:

~~a.~~ b At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes. X

~~b.~~ c At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by: X

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to (°F). X

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

d. ~~c~~. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

e. ~~d~~. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a control room essential filtration actuation, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4-inch Water Gauge relative to ~~the outside atmosphere~~ adjacent areas during system operation at a makeup flow rate to the control room of less than (1000 cfm).

AES 9/10/84

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a control room essential filtration actuation, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4-inch Water Gauge relative to ~~the outside atmosphere~~ adjacent areas during system operation at a makeup flow rate to the control room of less than (1000 cfm).

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PG 3/4 3-64, 65, 66 67	**CHANGES:**

MTEB

Add footnotes.

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68 **DELETE:**

ASB
MEB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is out belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-19, 4-20 **CHANGE:**

RSB

Tech. Spec. See page.

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 Typo.

PG 3/4 4-29 NEW TABLE WILL BE SUPPLIED IN A WEEK

MEB
RSB
MTEB

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.
   1. Four high pressure turbine stop valves.
   2. Four high pressure turbine control valves.
   3. Six low pressure turbine reheat stop valves.
   4. Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 20°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

THIS GRAPH WILL BE COMING IN

ABOUT A WEEK.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	38°	1.5	8 - 10
2	43°	1.5	Standby
3	137°	1.5	4 - 5
4	142°	1.5	Standby
5	230°	1.5	12 - 15
6	310°	1.5	18 - 24

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SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure ~~. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.~~

BUT BECAUSE OF HIGHER NEUTRON FLUENCE
MAY BE CONTROLLING

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp. ~~therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined.~~ Consequently, ~~for the cases in which the outer wall of the vessel becomes the stress controlling location,~~ each heatup rate of interest must be analyzed on an individual basis

FOR BOTH THE INNER AND OUTER WALL

FOR REVIEW

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and ~~copper content of the material in question~~, RESIDUAL ELEMENT CONTENT can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

TABLE 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a)(b) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_V ENERG FOR LONGITUDI DIRECTION-ft l
128-101	M-6703-1	SA 508-CL2	Inlet Nozzle	-20	0	+20	+60	N.A.
128-101	M-6703-2	SA 508-CL2	Inlet Nozzle	+10	+10	-25	+10	N.A.
128-101	M-6703-3	SA 508-CL2	Inlet Nozzle	-10	-10	-27	+18	N.A.
128-101	M-6703-4	SA 508-CL2	Inlet Nozzle	0	0	+5	+42	N.A.
131-102	M-4307-1	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
131-102	M-4307-2	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
128-501	M-6708-1	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-2	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-3	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-501	M-6708-4	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-301	M-4304-1	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
128-301	M-4304-2	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
131-101	M-6712-1	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-2	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-3	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
131-101	M-6712-4	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
126-101	M-6705-1	SA 508-CL2	Vessel Flange	-70	-70	-78	-28	N.A.
106-101	M-6706-1	SA 508-CL2	Closure Head Flange	-70	-70	-80	-54	N.A.

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values.
** = Average of three test results.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

(PLATES)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_V ENERGY FOR LONGITUDINAL DIRECTION-ft lb
142-102	M-4311-1	SA 533-GRB-CL1	Lower Shell Plate	-10	-10	-6	+40	134
142-102	M-4311-2	SA 533-GRB-CL1	Lower Shell Plate	-40	-40	-24	-8	127
142-102	M-4311-3	SA 533-GRB-CL1	Lower Shell Plate	-20	-20	-7	+14	142
124-102	M-6701-1	SA 533-GRB-CL1	Intermed. Shell Plate	-40	+30	+44	+90	83
124-102	M-6701-2	SA 533-GRB-CL1	Intermed. Shell Plate	-50	+40	+56	+98	96
124-102	M-6701-3	SA 533-GRB-CL1	Intermed. Shell Plate	-30	+40	+39	+89	100
122-102	M-6701-4	SA 533-GRB-CL1	Upper Shell Plate	-30	+60	+82	+120	N.A.
122-102	M-6701-5	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+49	+98	N.A.
122-102	M-6701-6	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+42	+96	N.A.
102-102A	M-6709-1	SA 533-GRB-CL1	Closure Head Dome	-20	+10	+36	+66	N.A.
102-102B	M-6709-2	SA 533-GRB-CL1	Closure Head Dome	-70	-20	+4	+37	N.A.
150-102	M-6715-1	SA 533-GRB-CL1	Bottom Head Dome	-30	-30	+2	+30	N.A.
150-102	M-6715-2	SA 533-GRB-CL1	Bottom Head Dome	-40	-10	+26	+50	N.A.

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.

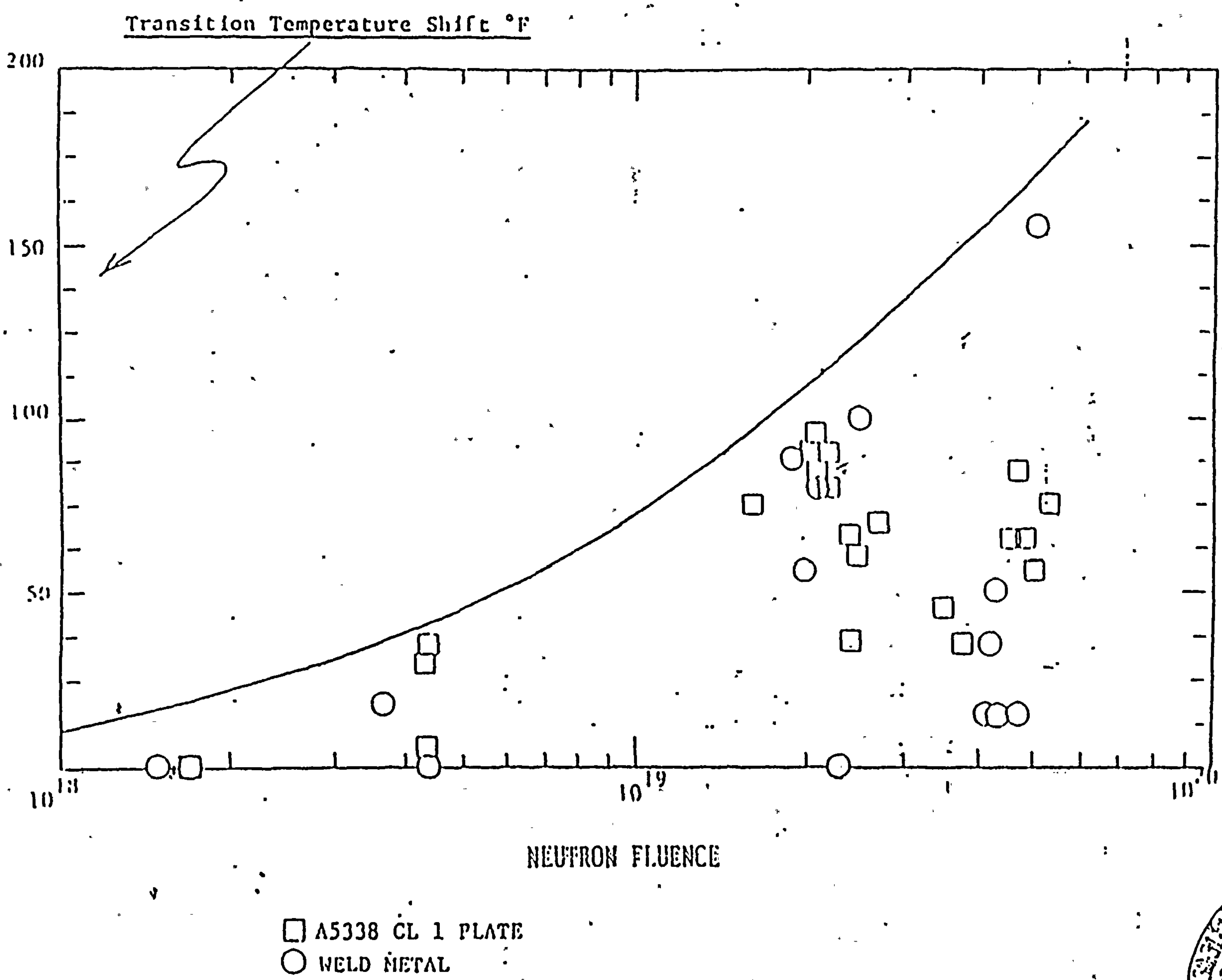

FIGURE B 3/4.4-1

NIL-DUCTILITY TRANSITION TEMPERATURE INCREASE AS A FUNCTION OF FAST (E > 1 MeV) NEUTRON FLUENCE (550°F IRRADIATION)

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~241~~ 255°F during cooldown and ~~201~~ 295°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

Materials Engineering Branch
draft comments from the reviewer

Add the following to T/S 4.4.8.1.2
to agree with Section 5.3.1 of the
Palo Verde SER.

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3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 20°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2 ← insert

X

4.4.8.1.2/4-28

Insert

~~operating limits~~ based on the greater of the following:

(1) the actual shift in reference temperature for plates M-6701-1 and M-4311-1 and weld 101-142 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld ~~101-171~~ and plate ~~M-6701-2~~ as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 1.15 seconds
2. Pressurizer Pressure - Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ ~~0.75~~ second 0.65
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second* 0.75
b. CEA Positions	≤ ~~0.22~~ second** 1.35
c. Cold Leg Temperature	≤ ~~0.81~~ second## 0.75
d. Hot Leg Temperature	≤ ~~0.81~~ second## 0.75
e. Primary Coolant Pump Shaft Speed	≤ ~~0.52~~ second# 0.75
f. Reactor Coolant Pressure from Pressurizer	≤ ~~0.48~~ second### 0.75
g. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second** 0.75
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 1.15 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

[#Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown.]

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 13 seconds. Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accomodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

THE PULSE TRANSMITTERS MEASURING PUMP SPEED ARE EXEMPT FROM RESPONSE TIME TESTING. THE RESPONSE TIME SHALL BE MEASURED FROM THE PULSE SHAPE INPUT

PROOF AND REVIEW

Marked up Proof
+ Review to be
Applicants Proposed

SECTION 6.0

ADMINISTRATIVE CONTROLS

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PROOF AND REVIEW

6.1 RESPONSIBILITY

6.1.1 The Director of Nuclear Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated SRO, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1. OR AS specified IN THE FSAR.

UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 OR AS specified IN THE FSAR and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the ~~(2)~~ other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (THIS EXCLUDES THE STA WORKING HOURS):

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the Director of Nuclear Operations or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Director of Nuclear Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

PROOF AND REVIEW

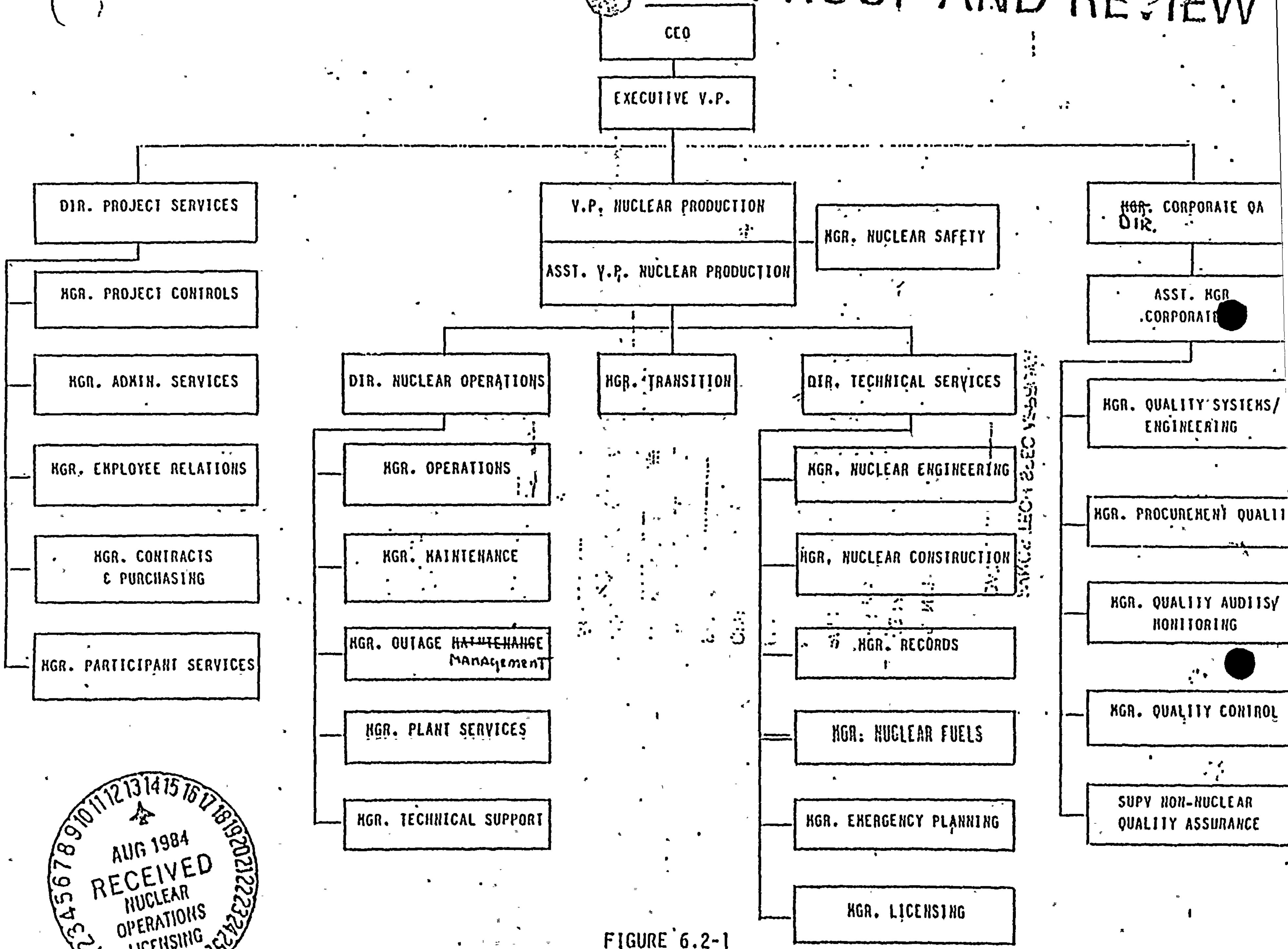

FIGURE 6.2-1
OFFSITE ORGANIZATION

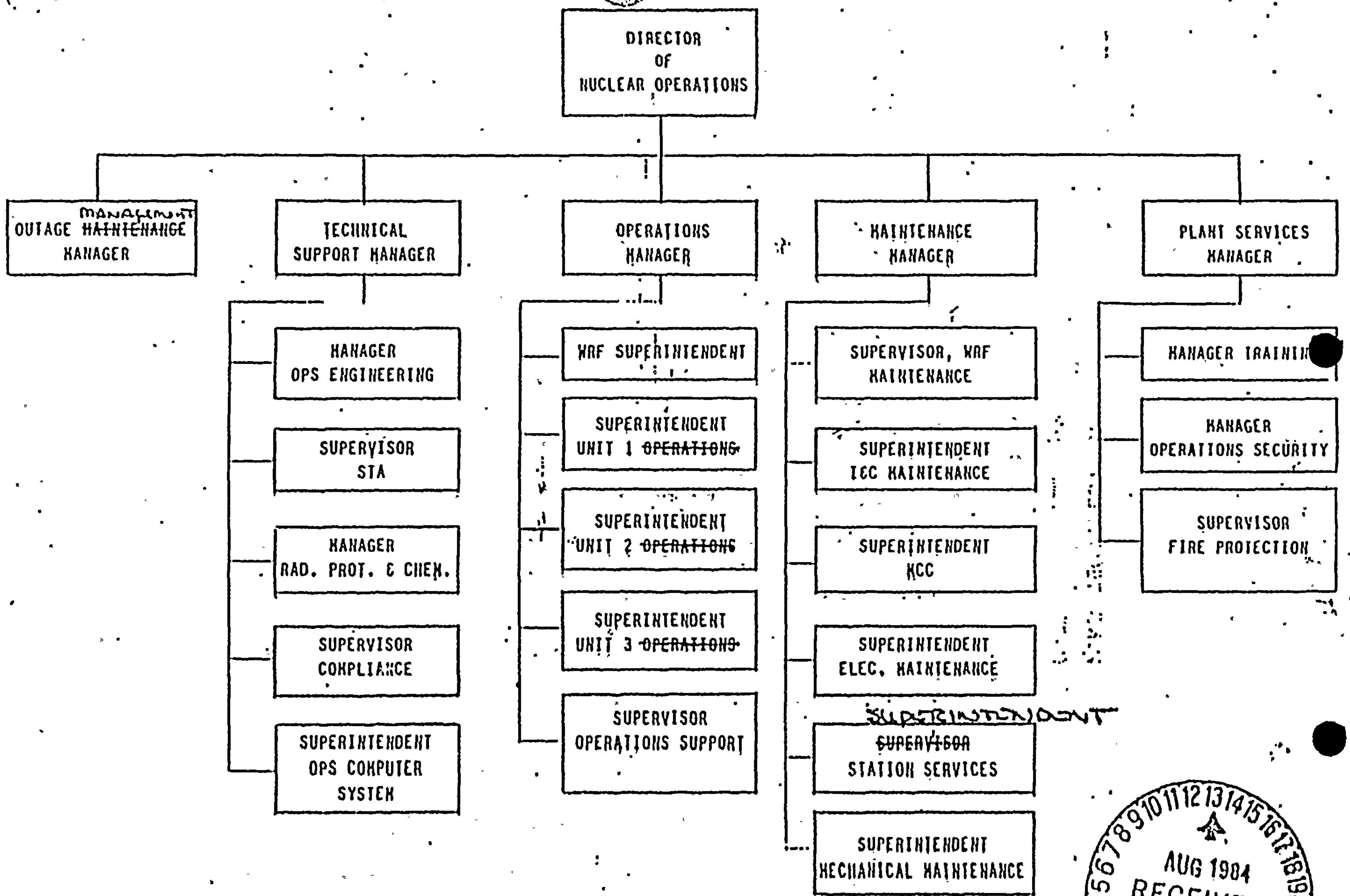

FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

PROOF AND REVIEW

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

PROOF AND REVIEW

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 ~~The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.~~

STET

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, Director of Nuclear Operations, and the Supervisor, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety. ~~and Supervisor of the NSG.~~

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978, as endorsed by Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director of Nuclear Operations or his designee and shall meet or exceed the requirements and recommendations of Section 5.5 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees; and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Director of Nuclear Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

Member:	Technical Support Manager
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Plant Services Manager
Member:	Engineering Manager
Member:	U1, U-2, U-3 ~~Operations~~ Superintendents
Member:	STA Supervisor
Member:	~~Training Manager~~ T AND C Superintendent
Member:	Radiation Protection and Chemistry Manager
Member:	Quality Systems/Engineering Manager

The Director of Nuclear Operations shall designate the Chairman and Vice-Chairman in writing.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairman, or his designated alternate.

QUORUM minimum

6.5.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice-Chairman, or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

Administrative control

g. a. Review of ~~(1)~~ all procedures ~~required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Director of Nuclear Operations, or designated alternate that affect Nuclear safety.~~ and changes

~~b. Review of all proposed tests and experiments that affect nuclear safety.~~

h. c. Review of all proposed changes to Appendix "A" Technical Specifications.

~~d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.~~

a. e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the ~~Director of Nuclear Operations, or designated alternate and to the~~ Nuclear Safety Group (NSG).

b. f. Review of ~~all REPORTABLE~~ EVENTS requiring 24-hour written notification to the Commission.

c. g. Review of unit operations to detect potential nuclear safety hazards.

d. h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Director of Nuclear Operations. ~~or designated alternate or the NSG.~~

~~i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

~~j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

f. k. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

RESPONSIBILITIES (Continued)

e. ~~l.~~ Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

~~m. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Director of Nuclear Operations and to the NSG.~~

~~n. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and radwaste treatment systems.~~

AUTHORITY

6.5.1.7 The PRB shall:

~~a. Recommend in writing to the Director of Nuclear Operations, or his designated alternate approval or disapproval of items considered under Specification 6.5.1.6a. through d. above.~~

a. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. ~~through e.~~ above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President-Nuclear Production, DNO and ~~the Supervisor of the~~ NSG of disagreement between the PRB and the Director of Nuclear Operations; however, the Director of Nuclear Operations ~~, or his designated alternate~~ shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1. above.

RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to ~~the Vice President Nuclear Production and the Supervisor of~~ the NSG.

COMMITTEES

6.5.1.9 Except for the review of programs and administrative control procedures, the PRB may establish standing committees to conduct reviews and make recommendations regarding the responsibilities identified in Specifications 6.5.1.6a., b., d., i., j., and 6.5.1.7a. and b. The following conditions must be met for such a committee:

a. The committee chairman shall be a PRB member designated in Specification 6.5.1.2;

b. The committee members shall be designated in writing by the PRB chairman;

Delete

~~RECORDS~~

~~6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Nuclear Safety Group.~~

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Director Nuclear Operations (DNO) shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Manager of Technical Support or the Manger of Engineering as previously designated by the Director of Nuclear Ops. Test results shall be approved by the Director of Nuclear Operations or the Manager of Technical Support.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the DNO; or by the Manager, Technical Support as previously designated by the DNO.

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the DNO to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the DNO, the Manager Technical Support, the Manager Operations, or the Manager Maintenance.

6.5.2.6 Review of the station security program, and implementing procedures, and submittal of recommended changes shall be approved by the DNO and transmitted to the Vice President-Nuclear Production and to the NSG.

PROOF AND REVIEW

ACTIVITIES (Continued)

6.5.2.7 Review of the station emergency plan, and implementing procedures, and submittal of recommended changes shall be approved.

6.5.2.8 The DNO shall assure the performance of a review by a qualified individual/organizaiton of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence.

New

6.5.2.9 The DNO shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

~~6.5.3 NUCLEAR SAFETY GROUP (NSG)~~

FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

~~COMMITTEES (Continued)~~

c. ~~The responsibility, authority, and functions of the standing committees, including such matters as quorum requirements and documentation of committee activities shall be defined in administrative control procedures;~~

d. ~~A report of the activities of each committee shall be made to the PRB at least once per calendar month; and~~

e. ~~Any matters which cannot be resolved among committee members, or which the committee feels may involve a change to the technical specifications or an unreviewed safety question shall be forwarded to the PRB for review prior to implementation.~~

Delete

6.5.~~2~~3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.~~2~~3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

COMPOSITION

6.5.~~2~~3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.~~2~~3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

REVIEW

program and its implementation

6.5.~~2~~3.4 The NSG shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question; This review will be done by the audit of selected safety evaluations.

REVIEW (Continued)

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS requiring 24 hour notification;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the NSG or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

prepared monthly for the Manager, Nuclear Safety, who will

AUDITS (Continued)

~~i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.~~

~~j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.~~

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Manager of Nuclear Safety on those areas of responsibility specified in Specifications 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared it and maintained. Report of reviews and audits shall be distributed ~~monthly~~ to the Vice President-Nuclear Production, Director of Nuclear Operations~~, Manager of Nuclear Safety,~~ and to the management positions responsible for the areas audited.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS: Requiring 24 hour notification

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the ~~Supervisor of the NSG~~ Manager of Nuclear Safety and the Vice President-Nuclear Production

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production~~, the Supervisor of the NSG,~~ and Director of Nuclear Operations and Manager Nuclear Safety shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the Supervisor of the NSG and the Vice President-Nuclear Production within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5.1 and approved prior to implementation. Programs and administrative control procedures shall be approved by the Director of Nuclear Operations, or designated alternate. Implementing procedures shall be approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

PROCEDURES AND PROGRAMS (Continued)

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant ~~management~~ supervisory staff, at least one of whom ~~holds a Senior Reactor Operator's License~~ on the unit affected. i.e. the Shift Supervisor or Assistant Shift Supervisor

c. The change is documented, reviewed in accordance with Specification 6.5.1 and approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the NSG at least once per 24 months:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, ~~the post-accident sample return piping of the radioactive waste gas system~~, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program will be written and issued prior to system being declared operational which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

e. Post-accident Sampling

will be written and approved prior to system being declared operational

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencment of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

ANNUAL REPORTS (Continued)

and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps**

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive ~~liquid and~~ gaseous effluents released from the unit or station during the previous calendar year. ~~This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period.~~ All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. ~~The meteorological conditions concurrent with the time of release of radioactive materials in gaseous~~

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

~~effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. [For ORs: approximate and conservative approximate methods are acceptable.]~~ The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from ~~liquid and~~ gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous ~~and liquid~~ radioactive material released to the environs. X

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Records of reactor tests and experiments.

RECORD RETENTION (Continued)

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of PRB meetings ~~and of NSC activities~~.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of Specifications 6.5.2.8 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

*Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

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HIGH RADIATION AREA (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area~~ and the maximum allowable stay time for individuals in that area~~. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

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*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (Continued)

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

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*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in ~~liquid and~~ gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto; X

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in ~~liquid and~~ gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; X

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

PROOF AND

PG 3/4 3-64, 65, 66 67 — **CHANGES:**

Add footnotes.

~~DATES~~ METB

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68 — **DELETE:**

ASB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is out belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-~~19~~ 18 4-20 — **CHANGE:**

RSB ~~ASB~~ MEB

Tech. Spec. See page.

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 — Typo.

PG 3/4 4-29 — NEW TABLE WILL BE SUPPLIED IN A WEEK

MEB RSB MEB

PG 3/4 4-26

METB

MEB

PG 3/4 7-24 CHANGE:

Words in Item H. See pages.

JUSTIFICATION:

These changes were made to make the paragraph read correctly.

PG 3/4 7-28 CHANGE:

CEB

LCO and Surveillance Requirement. See pages.

JUSTIFICATION:

We need to add +10% to the 1500gpm flow rate. This +10% was identified in our startup program. This change is in compliance with our requirements of ANI. This has also been discussed with our NRC Reviewer, D. Kubicki.

The 23 feet 1 inch corresponds to the readings the operators will be responsible for obtaining. The 23 feet 1 inch is equal to approximately 300,000 gallons.

Add the statement ... "when required to be operable." There are some valves that are not to be operable for each flow path.

PG 3/4 7-33 DELETE:

CEB

Table 3.7-3.

JUSTIFICATION:

This Table in Proof/Review is not correct. A new corrected Table will be sent.

PG 3/4 7-35 CHANGE:

CEB

Surveillance 4.7.11.3.2.b.1. See page.

JUSTIFICATION:

We do not have fire door release mechanisms in PVNGS.

PG 3/4 7-38 CHANGE:

CEB

Table 3.7-4. See page.

JUSTIFICATION:

Add four more hose station locations. This will bring the spec closer to PNVGS design.

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of ~~2235~~ 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve or within 72 hours following a system response to an Engineered Safety Feature actuation signal.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* TESTING PER SPECIFICATION 4.4.5.2.2.d IS NOT APPLICABLE DUE TO POSITIVE INDICATION OF VALVE POSITION IN THE CONTROL ROOM

1. LEAKAGE RATES LESS THAN OR EQUAL TO 1.0 GPM ARE CONSIDERED ACCEPTABLE

2. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED ACCEPTABLE IF THE LATEST MEASURED RATE HAS NOT EXCEEDED THE RATE DETERMINED BY PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

3. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED UNACCEPTABLE IF THE LATEST MEASURED RATE EXCEEDED THE RATE DETERMINED BY THE PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

4. LEAKAGE RATES GREATER THAN 5.0 GPM ARE CONSIDERED UNACCEPTABLE.

PROOF AND REVIEW

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE	DESCRIPTION
1) SIV 237	LOOP 1A RC/SI CHECK
2) SIV 247	LOOP 1B RC/SI CHECK
3) SIV 217	LOOP 2A RC/SI CHECK
4) SIV 227	LOOP 2B RC/SI CHECK
5) SIV 235	LOOP 1A SIT CHECK
6) SIV 245	LOOP 1B SIT CHECK
7) SIV 215	LOOP 2A SIT CHECK
8) SIV 225	LOOP 2B SIT CHECK
9) SIV 542	LOOP 1A SI HEADER CHECK
10) SIV 543	LOOP 1B SI HEADER CHECK
11) SIV 540	LOOP 2A SI HEADER CHECK
12) SIV 541	LOOP 2B SI HEADER CHECK
13) SIV 522	LOOP 1 HP LONG TERM RECIRCULATION CHECK
14) SIV 523	LOOP 1 HP LONG TERM RECIRCULATION CHECK
15) SIV 532	LOOP 2 HP LONG TERM RECIRCULATION CHECK
16) SIV 533	LOOP 2 HP LONG TERM RECIRCULATION CHECK
17) UV 651 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
18) UV 652 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION
19) UV 653 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
20) UV 654 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) ma be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested.

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon

as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

OR

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

THESE

FUNCTIONAL

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

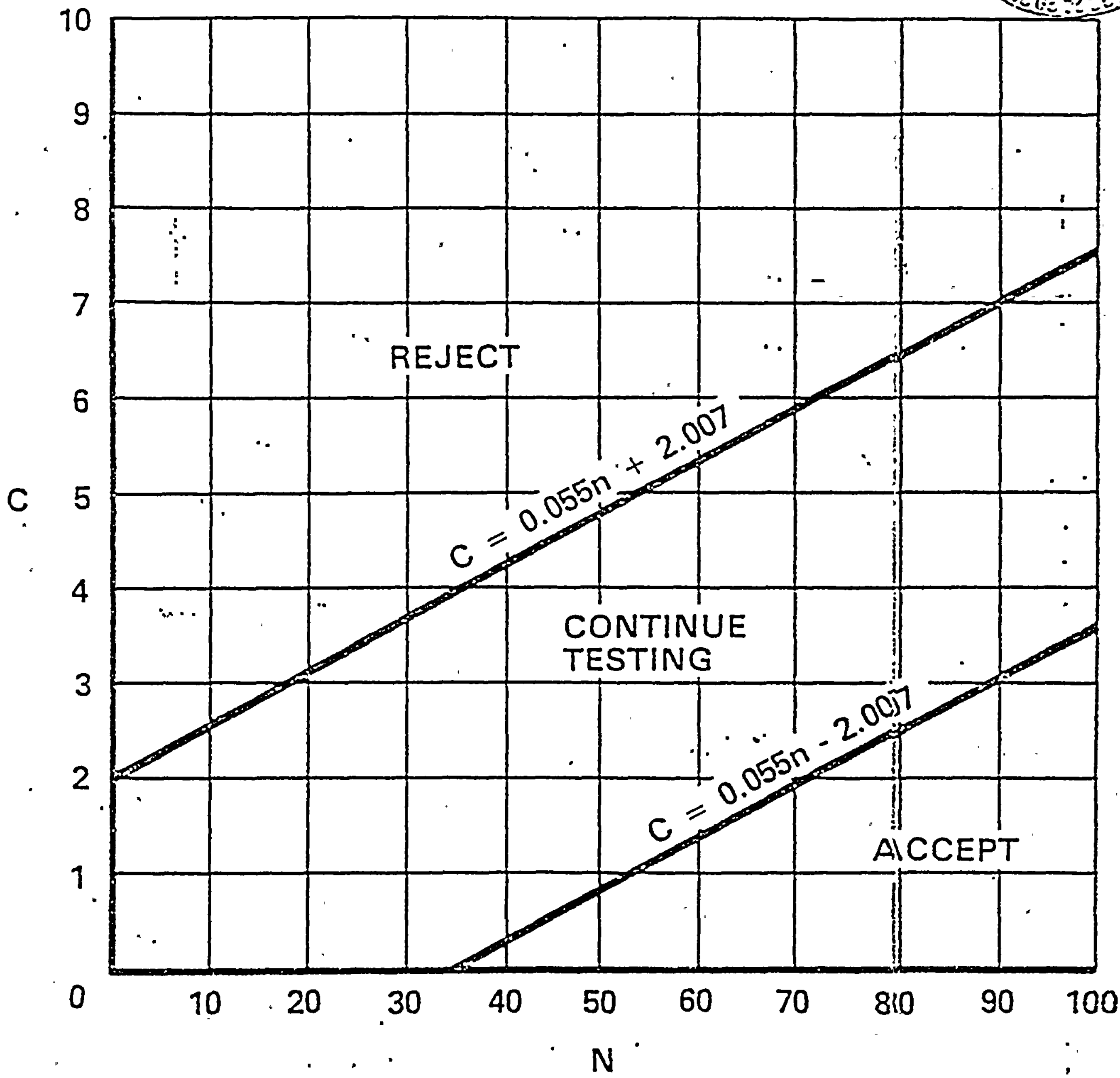

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1. except as noted by an asterisk (*).

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

* b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

* d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve or within 72 hours following a system response to an Engineered Safety Feature actuation signal.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* The provisions of Specification 4.4.5.2.2 (b) and (d) are not applicable for valves UV 651, UV 652, UV 653 and UV 654.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE	DESCRIPTION
1) SIV 237	LOOP 1A RC/SI CHECK
2) SIV 247	LOOP 1B RC/SI CHECK
3) SIV 217	LOOP 2A RC/SI CHECK
4) SIV 227	LOOP 2B RC/SI CHECK
5) SIV 235	LOOP 1A SIT CHECK
6) SIV 245	LOOP 1B SIT CHECK
7) SIV 215	LOOP 2A SIT CHECK
8) SIV 225	LOOP 2B SIT CHECK
9) SIV 542	LOOP 1A SI HEADER CHECK
10) SIV 543	LOOP 1B SI HEADER CHECK
11) SIV 540	LOOP 2A SI HEADER CHECK
12) SIV 541	LOOP 2B SI HEADER CHECK
13) SIV 522	LOOP 1 HP LONG TERM RECIRCULATION CHECK
14) SIV 523	LOOP 1 HP LONG TERM RECIRCULATION CHECK
15) SIV 532	LOOP 2 HP LONG TERM RECIRCULATION CHECK
16) SIV 533	LOOP 2 HP LONG TERM RECIRCULATION CHECK
* 17) UV 651	LOOP 1 SHUTDOWN COOLING ISOLATION
* 18) UV 652	LOOP 2 SHUTDOWN COOLING ISOLATION
* 19) UV 653	LOOP 1 SHUTDOWN COOLING ISOLATION
* 20) UV 654	LOOP 2 SHUTDOWN COOLING ISOLATION

* For these valves,

Leak rates greater than 1 gpm but less than or equal to 5.0 gpm will be acceptable provided, for each subsequent test, the latest measured leak rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater. Leakage rates greater than 5.0 gpm are unacceptable. ~~In addition, leak tests shall be performed after~~

PLANT SYSTEMS

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type ~~on any system~~ are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that ~~system~~ type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given ~~system~~ type shall be performed in accordance with the following schedule:

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SURVEILLANCE REQUIREMENTS (Continued)

No. of Inoperable Snubbers of Each Type ~~on Any System~~ per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. Insert All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. ~~For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.~~

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

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Insert

When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction.

SSPB

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SURVEILLANCE REQUIREMENTS (Continued)

as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers ~~where required~~, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

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power level in the reactor (decrease in coolant temperature and pressure) may aggravate the problem by inducing degassification of the primary coolant.

We are taking fuel building Ventilation exhaust and condenser vacuum plump/gland seal monitor out of Table 4/3-3 because they are duplicated in Table 3.3-13.

PG 3/4 3-39 CHANGE:

LPB

Surveillance 4.3.3.2b ~~and B~~

4.3.3.2.a not acceptable because it would allow not testing just after more detectors returned to service because they were tested 7 days earlier

JUSTIFICATION:

By adding the "7 days or more have elapsed since the last use" provides a time period for guidance for a channel check. This addition is consistent with SONGS tech. specs. They said without this clarification that there were problems with the region and plant interputation.

Delete last sentence of 4.3.3.2.b. We have no means to perform the calibration of the incore detectors. This is a function being performed by all supplfers of incores. Other utilities that have this Tech. Spec. say they cannot really meet this Spec. See justification for Page 3-64-67 on Page 15 for *** justification. —

PG 3/4 3-41 CHANGE:

SGEB

3.a (setpoint 0.02g).

JUSTIFICATION:

'FSAR Section 3.7.4.3, pg. 3.7-32.

PG 3/4 3-44 CHANGE:

~~AFEB~~ METB

1a and 1b.

JUSTIFICATION:

NUS recently supplied information showed the instrument range to be:

1 to 50 mph
1 to 50 mps

PG 3/4 3-52 CHANGE:

The LCO and Applicability. See page.

CEB

add "Fire" to FPER "Fire" Zone in Table -14- 3.3-11

CSB

The SDCHX to Containment Spray check valves are not within the scope of LCO 3.6.1.1 because they are required open during accident conditions.

The Safety Injection Long Term Recirculation check valves are not in the scope of LCO 3.6.1.1 because they are required to be open under accident conditions.

The remaining check valves in Table 3.6-1 are within the scope of LCO 3.6.1.1 and are now included in Table 3.6-0 of Tech. Spec. 3/4.6.1.1.

H. Footnote OPERABLE in LCO 3.6.3 with "A closed, isolated, or blank flanged valve is considered OPERABLE for containment isolation."

This note will clarify the intent of containment integrity as defined in the Tech. Spec. definitions and Tech. Spec. 3.6.1.1. The present wording of Tech. Spec. 3.6.3 could be misinterpreted as meaning that a normally closed and required closed CIAS/CPIAS valve that is found to be inoperable in the closed position violates LCO 3.6.3.

PG 3/4 6-3 <u>CHANGE:</u> CSB

Item B. See page.

<u>JUSTIFICATION:</u>

Adding ... At least "refueling or" every 18 months, "which ever comes first, " ...

Needs to be added to bring this statement into compliance with Appendix J of 10CFR.

PG 3/4 6-8 <u>CHANGE:</u>

~~CSB~~ SGEB

Action Item. See page.

<u>JUSTIFICATION:</u>

The above added phrase was added from an operating utility who helped us review our Tech. Spec. This comment was added and approved for them and helped them out. This added statement allows an engineering evaluation to be performed to identify if you have a structural problem or not before having to take any correction actions. The way the old Spec read you had to take action to restore containment structure integrity under the assumption that a problem existed. If no problem existed than corrective action was being taken without knowing the hole story.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Accelerometers	
a. Tendon Gallery Floor, 55' level	1
b. R.C.P., Upper Support, 129'6" level	1
c. Steam Generator Base, 101'9" level	1
d. Control Building Floor, 74' level	1
e. Auxiliary Building Floor 40' level	1
f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.)	1
2. Peak Reading Accelerograph	
a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level	1
3. Seismic Triggers	
a. Tendon Gallery Floor, 55' level (Setpoint ~~0.010 g)~~ 0.02g)	1
b. Containment Operating Floor, 140' level (Setpoint 0.010 g)	1
4. Digital Cassette Recorders	
a. Control Room Area, 140' level	1
b. Control Room Area, 140' level	1
c. Control Room Area, 140' level	1
d. Control Room Area, 140' level	1
e. Control Room Area, 140' level	1
f. Control Room Area, 140' level	1
5. Seismic Switches	
a. Tendon Gallery Floor, 55' level	1

	Horizontal	Vertical
Setpoint OBE	0.18 g	0.17 g
Setpoint SSE	0.31 g	0.34 g

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	[illegible] TEST
1. Triaxial Accelerometers			
a. Tendon Gallery Floor, 55' level	N.A.	R	SA
b. R.C.P., Upper Support, 129'6" level	N.A.	R	SA
c. Steam Generator Base, 101'9" level	N.A.	R	SA
d. Control Building Floor, 74' level	N.A.	R	SA
e. Auxiliary Building Floor 40' level	N.A.	R	SA
f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.)	N.A.	R	SA
2. Peak Reading Accelerograph			
a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level	N.A.	R	NA
3. Seismic Triggers			
a. Tendon Gallery Floor, 55' level	N.A.	R	SA
b. Containment Operating Floor, 140' level	N.A.	R	SA
4. Digital Cassette Recorders			
a. Control Room Area, 140' level	M	R	SA
b. Control Room Area, 140' level	M	R	SA
c. Control Room Area, 140' level	M	R	SA
d. Control Room Area, 140' level	M	R	SA
e. Control Room Area, 140' level	M	R	SA
f. Control Room Area, 140' level	M	R	SA
5. Seismic Switches			
a. Tendon Gallery Floor, 55' level	M	R	SA

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, ~~restore the structural integrity to within the limits within 24 hours or~~ be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

PERFORM AN ENGINEERING EVALUATION OF THE CONTAINMENT TO DEMONSTRATE ITS STRUCTURAL INTEGRITY WITHIN 72 HOURS; OTHERWISE

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Tendons The containment tendons' structural integrity shall be demonstrated at the end of 1, 3 and 5 years following the initial containment structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections the maximum first year lift off forces shall be decreased by the amount X log t, and the minimum lift off forces shall be decreased by the amount Y log t where t is the time interval in years from initial tensioning of the tendon to the current testing date. If no abnormal degradation of the containment tendons is detected during the 1-, 3-, and 5-year inspections, the number of tendons checked for lift-off force may be reduced to 2% of the population of each group (hoop and inverted U) with a minimum of three tendons fc each group. The sample size from any group need not exceed five. For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (hoop and inverted U) may be kept unchanged after the initial selection.

b. Removing one wire from one U tendon and one hoop tendon checked for lift-off force and determining that over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks, and damage.

2. There are no changes in the presence or physical appearance of the sheathing filler grease.

3. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.6.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.4 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

TABLE 4.6-1

TENDON SURVEILLANCE - FIRST YEAR

Tendon No.	Visual Inspection	Monitor Forces	Detension Tendon	Remove Wire	Test Wire
V32	X	X	No	No	No
V43	X	X	No	No	No
V62	X	X	X	X	X
V75*	X	X	A	A	A
H13-007*	X	X	X	X	X
H13-021	X	X	No	No	No
H21-037	X	X	No	No	No
H21-044	X	X	No	No	No
H32-016	X	X	No	No	No
H32-030	X	X	A	A	A

Notes:

1. "X" means the tendon shown shall be inspected for the stated requirements during this surveillance.
2. "A" means the tendon shown shall be inspected for the stated requirements during the next or second surveillance.
3. "No" means that inspection is not required for that temdon.
4. "*" means control tendon.

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TABLE 4.6-2

TENDON LIFT-OFF FORCE
U-TENDONS

PROOF AND REVIEW

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	Y
V3	Shop	1425	1304	22.9	32.4
	Field	1402	1282	22.7	32.2
V8	Shop	1513	1388	23.1	32.8
	Field	1466	1344	22.9	32.5
V11	Shop	1460	1338	22.9	32.5
	Field	1438	1317	22.9	32.5
V14	Shop	1525	1399	23.3	32.9
	Field	1491	1366	23.1	32.7
V15	Shop	1459	1337	22.9	32.5
	Field	1436	1315	22.8	32.3
V18	Shop	1476	1352	23.0	32.6
	Field	1453	1330	22.9	32.5
V21	Shop	1509	1385	23.0	32.6
	Field	1462	1341	22.8	32.3
V26	Shop	1402	1283	22.7	32.1
	Field	1471	1348	23.0	32.6
V29	Shop	1533	1408	23.1	32.8
	Field	1533	1408	23.1	32.8
V32	Shop	1463	1343	22.8	32.3
	Field	1510	1386	23.0	32.6
V36	Shop	1475	1354	22.9	32.4
	Field	1533	1408	23.1	32.8
V37	Shop	1510	1386	23.0	32.6
	Field	1486	1364	22.9	32.5
V40	Shop	1543	1416	23.3	32.9
	Field	1472	1350	22.9	32.5
V45	Shop	1457	1336	22.9	32.4
	Field	1526	1400	23.2	32.9
V47	Shop	1437	1315	22.9	32.5
	Field	1506	1379	23.3	32.9
V53	Shop	1466	1344	22.9	32.5
	Field	1513	1388	23.1	32.8
V54	Shop	1514	1389	23.1	32.8
	Field	1445	1324	22.8	32.3
V55	Shop	1457	1335	22.9	32.5
	Field	1469	1346	23.0	32.5
V62	Shop	1475	1354	22.9	32.4
	Field	1486	1364	22.9	32.5
V67	Shop	1442	1320	22.9	32.4
	Field	1476	1352	23.0	32.6
V72	Shop	1510	1386	23.0	32.6
	Field	1510	1386	23.0	32.6
V75	Shop	1527	1402	23.1	32.8
	Field	1504	1380	23.0	32.6
V83	Shop	1458	1334	23.0	32.6
	Field	1389	1270	22.7	32.1
V86	Shop	1505	1378	23.3	32.9
	Field	1436	1314	22.9	32.5

TABLE 4.6-2 (Continued)

TENDON LIFT-OFF FORCE
HOOP TENDONS

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	
H21-001	Shop	1516	1384	29.8	42.3
	Field	1505	1373	29.8	42.3
H21-003	Shop	1487	1356	29.7	42.2
	Field	1499	1367	29.8	42.3
H21-006	Shop	1422	1292	29.6	42.0
	Field	1513	1377	30.1	42.7
H13-008	Shop	1441	1309	29.7	42.2
	Field	1475	1341	29.9	42.4
H32-009	Shop	1517	1382	30.0	42.5
	Field	1493	1360	29.8	42.3
H32-010	Shop	1480	1345	30.0	42.5
	Field	1480	1345	30.0	42.5
H21-011	Shop	1505	1371	29.9	42.4
	Field	1458	1328	29.7	42.1
H32-012	Shop	1434	1303	29.7	42.2
	Field	1480	1345	30.0	42.5
H21-013	Shop	1470	1339	29.7	42.2
	Field	1470	1339	29.7	42.2
H13-014	Shop	1473	1339	30.0	42.5
	Field	1450	1317	29.8	42.3
H32-015	Shop	1517	1382	30.0	42.5
	Field	1505	1371	29.9	42.4
H32-016	Shop	1411	1282	29.6	42.0
	Field	1457	1324	29.8	42.3
H13-019	Shop	1445	1315	29.6	42.0
	Field	1445	1315	29.6	42.0
H13-021	Shop	1515	1380	30.0	42.5
	Field	1491	1358	29.8	42.3
H21-021	Shop	1445	1315	29.6	42.0
	Field	1445	1315	29.6	42.0
H32-023	Shop	1470	1339	29.7	42.2
	Field	1493	1360	29.8	42.3
H13-025	Shop	1412	1284	29.4	41.8
	Field	1505	1371	29.9	42.4
H32-026	Shop	1389	1261	29.5	41.9
	Field	1480	1345	30.0	42.5
H21-028	Shop	1473	1339	30.0	42.5
	Field	1450	1317	29.8	42.3
H13-030	Shop	1428	1297	29.7	42.2
	Field	1428	1297	29.7	42.2
H32-033	Shop	1470	1339	29.7	42.2
	Field	1505	1371	29.9	42.4
H21-037	Shop	1505	1371	29.9	42.4
	Field	1446	1317	29.6	42.0
H32-040	Shop	1417	1286	29.7	42.1
	Field	1417	1286	29.7	42.1

TABLE 4.6-2 (Continued)

TENDON LIFT-OFF FORCE
HOOP TENDONS

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	Y
H32-041	Shop	1475	1352	23.0	32.5
	Field	1522	1395	23.2	32.9
H21-042	Shop	1507	1381	23.1	32.8
	Field	1530	1403	23.3	32.9
H13-043	Shop	1421	1302	22.7	32.1
	Field	1492	1368	23.0	32.6
H32-044	Shop	1472	1349	23.0	32.5
	Field	1530	1403	23.3	32.9
H21-045	Shop	1510	1385	23.1	32.7
	Field	1545	1418	23.3	32.9
H21-046	Shop	1537	1410	23.3	32.9
	Field	1548	1421	23.3	33.0
H13-047	Shop	1439	1319	22.7	32.2
	Field	1486	1363	23.0	32.5

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Accelerometers			
a. Tendon Gallery Floor, 55' level	~~N.A.~~ M	R	SA
b. R.C.P., Upper Support, 129'6" level	~~N.A.~~ M	R	SA
c. Steam Generator Base, 101'9" level	~~N.A.~~ M	R	SA
d. Control Building Floor, 74' level	~~N.A.~~ M	R	SA
e. Auxiliary Building Floor 40' level	~~N.A.~~ M	R	SA
f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.)	~~N.A.~~ M	R	SA
2. Peak Reading Accelerograph			
a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level	N.A.	R	NA
3. Seismic Triggers			
a. Tendon Gallery Floor, 55' level	~~N.A.~~ M	R	SA
b. Containment Operating Floor, 140' level	~~N.A.~~ M	R	SA
4. Digital Cassette Recorders			
a. Control Room Area, 140' level	M	R	SA
b. Control Room Area, 140' level	M	R	SA
c. Control Room Area, 140' level	M	R	SA
d. Control Room Area, 140' level	M	R	SA
e. Control Room Area, 140' level	M	R	SA
f. Control Room Area, 140' level	M	R	SA
5. Seismic Switches			
a. Tendon Gallery Floor, 55' level	M	R	SA

note: changes from
SGEB/DE
9/17/84

ENCLOSURE 2
RECOMMENDED STS FOR PRESTRESSED CONCRETE
CONTAINMENT STRUCTURAL INTEGRITY

CONTAINMENTS SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY [Prestressed concrete containment with ungrouted tendons.]

LIMITING CONDITION FOR OPERATION

3.6.1.~~7~~6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.7 except for Specification 4.6.1.7.2a.4), restore the containment vessel to the required level of integrity within 15 days, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.7.2a.4), restore the containment vessel to the required level of integrity within 72 hours, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~7~~6.1 The structural integrity of the containment vessel shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. All of the acceptance testing of tendon and visual examinations of end anchorages, adjacent concrete surfaces and containment vessel surfaces shall be performed sequentially and within the same time frame.

4.6.1.~~7~~6.2 The structural integrity of the tendons shall be demonstrated by:

a. Determining from a random but representative sample of at least _______ tendons (_______ hoop, _______ vertical, _______ dome, and _______ inverted U) that each group ([dome, vertical, hoop, and inverted U]) has an observed lift-off force within the predicted limits for that group. For each subsequent inspection one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the

tendon acceptance criteria shall be as follows:

1) If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability;

2) If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence;

3) If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be performed to determine the cause and extent of such occurrence;

4) If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as below the acceptance criteria for containment vessel structural integrity; and

5) Unless there is degradation of the containment vessel below the acceptance criteria during the first three inspections, the sample population for subsequent inspections shall include at least ________ tendons (______ hoop, _______ vertical, _________ dome, and ______ inverted U).

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires. A previously stressed tendon wire or strands from one tendon of each group shall be removed for testing and examination over the entire length to determine (which should include the broken wire if so identified) that:

1) The tendon wires are free of corrosion, cracks, and damage;

2) There are no changes in the presence or physical appearance of the sheathing filler-grease; and

3) A minimum tensile strength of _____ psi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence that structural integrity is below the acceptance criteria.

c. Performing tendon retensioning of those tendons detensioned for inspection to at least force level recorded prior to detensioning or the predicted value, whichever is greater, with the tolerance within minus zero to plus 6%, except that the final seating force shall be such that the stress in the wire or strand shall not exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons, the stress in the tendon shall not exceed 80% of its ultimate strength, and the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific loads differs by more than 10% from that recorded during installation, an investigation shall be made to ensure that the difference is not related to wire failures or slips of wires in anchorages; and

d. Verifying the OPERABILITY of the sheathing filler-grease by assuring:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified as follows:

Water content	0 - 5% by wt.
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm
Sulfides	0 - 5 ppm
Reserved Alkalinity	0 - 50% of the installed value
(Base Numbers)	(installed value 0-5 for older grease).

6
4.6.1.~~7~~.3 As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. For those containments in multiple unit plants for which only visual inspection need be performed, tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load-bearing components of the anchorages. The surrounding concrete shall also be checked visually for indication of any abnormal condition.

6
4.6.1.~~7~~.4 The exterior surface of the containment vessel shall be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration of disintegration, or grease leakage, each of which can be considered as evidence that the structural integrity is below the acceptance criteria.

3/4.6.1.7 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

[Prestressed concrete containment with ungrouped tendons]

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of [48] psig in the event of a [LOCA or steam line break accident]. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 3, 1984, and Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," 1984.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

PROOF AND REVIEW

PG VI

CHANGE TO:

Pressurizer Heatup/Cooldown Limits . . . . 3/4 4-31 me
Overpressure Protection Systems 3/4 4-32
Structural Integrity 3/4 4-33

JUSTIFICATION:

Typo's in page numbers.

PG VII

CHANGE TO:

Hydrogen Monitors . . . . 3/4 6-36 me
Electric Hydrogen Recombiners 3/4 6-37
Hydrogen Purge Cleanup System 3/4 6-38

JUSTIFICATION:

Typo's in page numbers.

3/4 VIII

DELETE:

Halon Systems . . . . 3/4 7-36 CEB

JUSTIFICATION:

APS doesn't have Halon Systems and do not have the Tech Spec.

CHANGE:

Fire Hose Stations . . . . 3/4 7-37 me

JUSTIFICATION:

Type in page number.

PG IX

CHANGE:

Monitor Operated Thermal Overload Protection
and Bypass Devices . . . . 3/4 8-33 me
Water Level Storage Pool 3/4 9-13
Fuel Building Essential Ventilation System 3/4 9-14

JUSTIFICATION:

Typo's in page numbers.

3269D/0126D

PG XVI CHANGE:

Meeting Frequency 6-7

me

JUSTIFICATION:

Typo's in page numbers.

PG XIX CHANGE:

3.4-1 ...Power with the primary...

me

JUSTIFICATION:

Typo.

PG XXI CHANGE:

4.4-5 Reactor vessel material surveillance program withdrawal schedule 3/4 4-30

4.6-1 Tendon surveillance 3/4 6-10

me

PG XXII 3.8-3 Motor-operated valves 3/4 8-34

JUSTIFICATION:

Typo's.

PG 1-2 CHANGE 1.7.9.2 TO:

...provided in table 3.6-0 of specification 3.6.1.1

CSB

JUSTIFICATION:

To reference the proper correct table and specification number.

B3/4.3-3

PG ~~1-7~~ ADD ~~1.3.9~~ paragraph

Fire Protection Evaluation Report as discussed in the PVNGS FSAR.

CEB

JUSTIFICATION:

Tech Spec 3.3.3.7 refers to a fire detection zone. PVNGS uses the FPER zone terminology instead of a fire detection zone. This change is consistent with PVNGS Operations Terminology and FSAR.

power level in the reactor (decrease in coolant temperature and pressure) may aggravate the problem by inducing degassification of the primary coolant.

We are taking fuel building Ventilation exhaust and condenser vacuum plump/gland seal monitor out of Table 4/3-3 because they are duplicated in Table 3.3-13.

PG 3/4 3-39 **CHANGE:**

Surveillance 4.3.3.2a and B.

LPB

JUSTIFICATION:

By adding the "7 days or more have elapsed since the last use" provides a time period for guidance for a channel check. This addition is consistent with SONGS tech. specs. They said without this clarification that there were problems with the region and plant interputation.

Delete last sentence of 4.3.3.2.b. We have no means to perform the calibration of the incore detectors. This is a function being performed by all suppliers of incores. Other utilities that have this Tech. Spec. say they cannot really meet this Spec. See justification for Page 3-64-67 on Page 15 for *** justification.

PG 3/4 3-41 **CHANGE:**

SGEB

3.a (setpoint 0.02g).

JUSTIFICATION:

FSAR Section 3.7.4.3, pg. 3.7-32.

PG 3/4 3-44 **CHANGE:**

METB

1a and 1b.

JUSTIFICATION:

NUS recently supplied information showed the instrument range to be:

1 to 50 mph
1 to 50 mps

PG 3/4 3-52 **CHANGE:**

The LCO and Applicability. See page.

CEB

PROOF

JUSTIFICATION:

PVNGS does not use "fire detection zone" terminology our fire detection equipment is located in the fire zone.

Change from Function A(B) to Function X(Y) is to be consistent with Table 3.3-11.

CEB

PG 3/4 ~~3-56~~ 53 to 3-56 CHANGE:

~~Note Typo's.~~ New Table 3.3-11

JUSTIFICATION:

~~Typo's.~~

PG 3/4 3-58 CHANGE:

Typo's, see page.

PG 3/4 3-60, 61, 62 CHANGE:

See pages.

~~DAGS~~ METB

JUSTIFICATION:

instead of * - this monitor is only needed during waste gas release; therefore, requiring this instrument to be operable all the time is not appropriate.

is used because this monitor is only needed during the conditions cited in the new note. There is no radiological hazard except only in these conditions.

#RU-144 has been deleted from Table 3.3-13 as the accident range noble gas monitors are required under commitments to NUREG-0737. The presence of RU-144 in this location causes confusion of the interpretation of the operability requirements of RU-143 and RU-144.

The present configuration of the Technical Specification Table 3.3-13 would lead one to believe that RU-144 would be adequate to replace the function of RU-143. This is not an appropriate interpretation. RU-144 is not in any way a redundant channel to RU-143 which is the normal range noble gas effluent monitor.

Note change Justification were identified above.

PG 3/4 3-63 CHANGES NOTED ON PAGE 13 OF THIS ATTACHMENT

~~DAGS~~ METB

PROOF AND

PG 3/4 7-24 **CHANGE:** MEB -

Words in Item H. See pages.

JUSTIFICATION:

These changes were made to make the paragraph read correctly.

PG 3/4 7-28 **CHANGE:** CEB

7-29

LCO and Surveillance Requirement. See pages.

JUSTIFICATION:

We need to add ±10% to the 1500gpm flow rate. This ±10% was identified in our startup program. This change is in compliance with our requirements of ANI. This has also been discussed with our NRC Reviewer, D. Kubicki.

The 23 feet 1 inch corresponds to the readings the operators will be responsible for obtaining. The 23 feet 1 inch is equal to approximately 300,000 gallons.

Add the statement ... "when required to be operable." There are some valves that are not to be operable for each flow path.

PG 3/4 7-33 **DELETE:** CEB

Table 3.7-3.

JUSTIFICATION:

This Table in Proof/Review is not correct. A new corrected Table ~~will be sent.~~ is enclosed.

PG 3/4 7-35 **CHANGE:** CEB

Surveillance 4.7.11.3.2.b.1. See page.

JUSTIFICATION:

We do not have fire door release mechanisms in PVNGS.

PG 3/4 7-38 **CHANGE:** CEB

Table 3.7-4. See page.

JUSTIFICATION:

Add four more hose station locations. This will bring the spec closer to PNVGS design.

PG 3/4 7-39 CHANGE: CEB

Surveillance 4.7.11.6.b. See page.

JUSTIFICATION:

Delete ... "and verifying that the hydrant barrel is dry "... This Spec is used for those plants in climates where freezing occurs. PVNGS, as discussed in the FSAR, is not subject to climates or weather that would cause water in the hydrant barrel from freezing and causing various damage.

PG 3/4 7-41 CHANGE: CEB

LCO and Surveillance 4.7.12.1.b. See page.

JUSTIFICATION:

Delete the fire windows reference. PVNGS doesn't have fire windows.

PG 3/4 7-42 CHANGE: CEB

Surveillance 4.7.12.2. See page.

JUSTIFICATION:

Deletion of Item A is justified in that PVNGS doesn't have any fire door supervision system as described; therefore, we cannot comply with this.

PG 3/4 7-43 CHANGE:

Surveillance 4.7.13.b. See page. RSB

JUSTIFICATION:

CE new numbers.

PG 3/4 8-1 CHANGE:
8-2
8-7

Tech. Spec. See pages. PSB ORAB

JUSTIFICATION:

The Tech. Spec. was changed to comply with NRC Generic letter 84-15.

Technical Specification 3/4.8.1, "A.C. Sources"

The proposed changes to Tech. Spec. 3/4.8.1 are a result of the applicable recommendations of NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability" dated July 2, 1984.

Pg	Branch(s)	Pg	Branch(s)
B 3/4 1-1	CPB		
B 3/4 1-2	CPB	B3/4 7-1	RSB ASB
1-3	CPB	7-2	RSB ASB
2-1	CPB		
3-3	CEB	7-3	ASB
3-4	ASB	7-4	MEB
4-1	RSB	7-6/7	CEB
4-6	RSB	9-3	AEB ASB
4-7	RSB		
4-11	RSB	10-1	CPB
		10-2	RSB
5-1	CPB	11-1/2	RSB METB
5-3	RSB		
6-2	CSB	11-5	METB
6-3/4	AEB CSB		

PROOF AND REVIEW

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.38 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

FPER

1.39 FIRE PROTECTION EVALUATION REPORT AS DISCUSSED IN PVNGS FSAR

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each ~~fire detection~~ zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone Function A fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at ~~least~~ once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

PROOF AND REVIEW

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - CONTROL			
1	74'	Essential Chiller Rm. - Train A			24/0
2	74'	Essential Chiller Rm. - Train B			21/0
3	74'	Cable Shaft - Trains A & B			2/0
86	74-156'4"	Deadspace Compartment - Trains A & B	0/2		0/6
4	100'	Cable Shaft - Trains A & B			2/0
5	100'	ESF Switchgear Rooms - Trains A & B			0/20
6	100'	DC Equip. Rms. - Tr. A (Chan. C) Tr. B (Chan. D)			4/0
7	100'	DC Equip. Rms. - Tr. A (Chan. A) Tr. B (Chan. B)			4/0
8	100'	Battery Rms. - Tr. A (Chan. C) Tr. B (Chan. D)	0/4		0/4
9	100'	Battery Rms. - Tr. A (Chan. A) Tr. B (Chan. B)	0/4		0/4
10	100'	Remote Shutdown Rm.			2/0
11	120'	Cable Shafts - Tr. A & B			2/0
14	120'	Lower Cable Spreading Rm.	0/6		0/44
15	140'	Cable Shaft Tr. A & B			2/0
17	140'	Control Rm. - MCB's & Relay Cabinets			97/0
18	160'	Cable Shafts - Tr. A & B			2/0
20	160'	Upper Cable Spreading Rm.	0/5		0/44

See TAB C

PROOF AND REVIEW

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - DIESEL GENERATOR			
21	100'	Diesel Generator - Tr. A & B	0/6	0/8	
22	100'	Diesel Generator Control Rms. Tr. A & B			2/0
24	115'	Combustion Air Intake Rms. - Tr. A & B			2/0
23	131'	Fuel Oil Day Tanks - Tr. A & B	0/2		
25	131'	Exhaust Silencer Rm. - Tr. A & B		6/0	
		BUILDING - FUEL			
28	100'	Spent Fuel Pool Cooling and Cleanup Pump Areas			3/0
		BUILDING - AUXILIARY			
30	51'-6"	Containment Spray Pump Rms. - Tr. A & B			0/4
31	51'-6"	HPSI Pump Rms. - Tr. A & B			0/4
32	51'-6"	LPSI Pump Rms. - Tr. A & B			0/4
34	70'	ECW Pump Rms. - Tr. A & B			4/0
35	70'	Shutdown Cooling Ht. x Chgr. Tr. A & B			8/0
37	70'	Piping Penetration Rm. - Tr. A & B			4/0
37A	70'	Corridors - East & West			18/0
39	88'	Pipeways - Tr. A & B			25/0
2A	100'	Elect. Penetration Rm. Tr. A (Chan. C)	0/1		0/25

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
42B	100'	Elect. Penetration Rm. Tr. B (Chan. B)	0/1		0/24
42C	100'	Corridors - East & Southeast	0/1		3/35
42D	100'	Corridor - West	0/1		0/29
46	100'	Charging Pump and Valve Gallery Rms.			0/9
47A	120'	Elect. Penetration Rm. - Tr. A (Chan. A)	0/1		0/28
47B	120'	Elect. Penetration Rm. Tr. B (Chan. D)	0/1		0/24
48	120'	ECW Surge Tanks Corridor - Tr. A & B			0/2
52A	120'	Central Corridor - West	0/1		4/15
52D	120'	Central Corridor - East	0/1		4/17
54	120'	Reactor Trip Switchgear Rm.	1/0		6/0
		BUILDING - CONTAINMENT**			
66	100'	Containment Cable Tray Area - South	1/0		
67	100'	Containment Cable Tray Area - North	1/0		
66	120'	Containment Cable Tray Area - South	1/0		
67	120'	Containment Cable Tray Area - North	1/0		
68	120'	Steam Generator Areas North & South			12/0
69	120'	Containment Cable Tray Area - East	1/0		

See TABLE

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
67	140'	Containment Cable Tray Area - North	1/0		
68	140'	Steam Generator Area - North & South			10/0
70	140'	Cavity Cooling Fans			4/0
71	140'	Charcoal Filter Area - North & South			4/0
		MAIN STEAM SUPPORT STRUCTURE			
72	80'	Turbine Driven Aux. Feedpump Rm.			0/3
73	80'	Motor Driven Aux. Feedpump Rm.			1/1
74	100'	Main Steam Isol. & Dump Valve Area			0/4
74	120'	Main Steam Isol. & Dump Valve Area			0/4
74	140'	Main Steam Isol. & Dump Valve Area			0/4

* (x/y): x is the number of Function A (early warning fire detection and notification only) instruments.

y is the number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

PROOF AND REVIEW

TABLE 3.3-11

FIRE DETECTION INSTRUMENT

FPER ZONE FIRE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - CONTROL			
1	74'	Essential Chiller Rm. - Train A			24/0
2	74'	Essential Chiller Rm. - Train B			21/0
3A	74'	Cable Shaft - Train A			1/0
3B	74'	Cable Shaft - Train B			1/0
86A	74-156'4"	Deadspace Compartment - Train A	0/1		0/3
86B	74-156'4"	Deadspace Compartment - Train B	0/1		0/3
4A	100'	Cable Shaft - Train A			1/0
4B	100'	Cable Shaft - Train B			1/0
5A	100'	ESF Switchgear Room Train A			0/10
5B	100'	ESF Switchgear Room Train B			0/10
6A	100'	DC Equipment Rm. - Tr. A (Channel C)			2/0
6B	100'	DC Equipment Rm. - Tr. B (Channel D)			2/0
7A	100'	DC Equipment Rm. - Tr. A (Channel A)			2/0
7B	100'	DC Equipment Rm. - Tr. B (Channel B)			2/0
8A	100'	Battery Rm. - Tr. A (Channel C)	0/2		0/2
8B	100'	Battery Rm. - Tr. B (Channel D)	0/2		0/2

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
9A	100'	Battery Rm - Train A (Channel A)	0/2		0/2
9B	100'	Battery Rm - Train B (Channel B)	0/2		0/2
10A	100'	Remote Shutdown Rm. Train A			1/0
10B	100'	Remote Shutdown Rm. Train B			1/0
11A	120'	Cable Shaft - Train A			1/0
11B	120'	Cable Shaft - Train B			1/0
14	120'	Lower Cable Spreading Rm.			
		a. System 1	0/1		0/6
		b. System 2	0/1		0/6
		c. System 3	0/1		0/8
		d. System 4	0/1		0/8
		e. System 5	0/1		0/8
		f. System 6	0/1		0/8
15A	140'	Cable Shaft Tr. A			1/0
15B	140'	Cable Shaft Tr. B			1/0
17	140'	Control Rm - MCB's & Relay Cabinets			97/0
18A	160'	Cable Shaft - Train A			1/0
18B	160'	Cable Shaft - Train B			1/0

X

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PROOF AND REVIEW

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER (FIRE) ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
20	160'	Upper Cable Spreading Rm.			
		System 1	0/1		0/12
		System 2	0/1		0/8
		System 3	0/1		0/8
		System 4	0/1		0/8
		System 5	0/1		0/8
		BUILDING - DIESEL GENERATOR			
21A	100'	Diesel Generator - Tr. A	0/3	0/4	
21B	100'	Diesel Generator - Tr. B	0/3	0/4	
22A	100'	Diesel Generator Control Rm. Tr. A			1/0
22B	100'	Diesel Generator Control Rm. Tr. B			1/0
24A	115'	Combustion Air Intake Rm. Tr. A			1/0
24B	115'	Combustion Air Intake Rm. Tr. B			1/0
23A	131'	Fuel Oil Day Tank Tr. A	0/1		
23B	131'	Fuel Oil Day Tank Tr. B	0/1		
25A	131'	Exhaust Silencer Rm. Tr. A		3/0	
25B	131'	Exhaust Silencer Rm. Tr. B		3/0	
		BUILDING - FUEL			
28	100'	Spent Fuel Pool Cooling and Cleanup Pump Areas			3/0

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING Auxiliary			
88A	51'-6"	west Corridors			6/0
88B	51'-6"	Exit corridors			6/0
32A	51' - 6"	LPSI Pump Rm. - Tr. A			0/2
32B	51' - 6"	LPSI Pump Rm. - Tr. B			0/2
34A	70'	ECW Pump Rm. - Tr. A			2/0
34B	70'	ECW Pump Rm. - Tr. B			2/0
35A	70'	Shutdown Cooling Ht. x Chgr. Tr. A			4/0
35B	70'	Shutdown Cooling Ht. x Chgr. Tr. B			4/0
37C	70' +88'	Piping Penetration Rm. - Tr. A			5/0
37D	70' +88'	Piping Penetration Rm. - Tr. B			4/0
37B	70'	Corridors - East			11/0
37A	70'	Corridors - West			11/0
39A	88'	Pipeways - Tr. A			5/0
39B	88'	Pipeways - Tr. B			8/0
42A	100'	Elect. Penetration Rm. Tr. A (Channel C)	0/1		0/25

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
42B	100'	Elect. Penetration Rm. Tr. B (Channel B)	0/1		0/24
42C	100'	Corridors - East + Southeast	0/2		3/35
42D	100'	Corridor - west	0/1		2/29
46A	100'	Charging Pump + Valve Gallery Rm - TR. A			0/3
46B	100'	" " - TR. B			0/3
46E	100'	" " - TR. E			0/3
47A	120'	Elect. Penetration Rm. Tr. A (Channel A)	0/1		0/28
47B	120'	Elect. Penetration Rm. Tr. B (Channel D)	0/1		0/24
48	120'	ECW Surge Tanks Corridor - TR. A+B			3/0
50B	120'	Valve Gallery			1/0
51B	120'	Spray Chemical Storage Tk Rm			1/0
52A	120'	Central Corridor - west	0/1		5/17
52D	120'	Central Corridor - East	0/1		7/17
54	120'	Reactor Trip Switchgear Rm	1/0		6/0
56B	140'	Storage and Elect. Equip. Rm - East			6/0
57I	140'	Clothing Issue and Men's Locker Rm			5/0
57J	140'	Women's Locker, Clean Storage and Lunch Rms.			7/0
57N	140'	Corridor area			4/0

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		Building - Containment **			
66A+66B	100'	Southwest and Southeast Perimeter	1/0		
67A+67B	100'	northwest and Northeast Perimeter	1/0		
66A	120'	Southwest Perimeter	1/0		
66B	120'	Southeast Perimeter	1/0		
67A+67B	120'	northwest and Northeast Perimeter	1/0		
63A	120'	no. 1 RCPS and SG area			6/0
63B	120'	no. 2 RCPS and SG area			6/0
B, 67A+67B	140'	Southwest, Southeast, northwest and Northeast Perimeters	1/0		
63A	140'	no. 1 RCPS and SG area			5/0
63B	140'	no. 2 RCPS and SG area			5/0
70	140'	Refueling Pool and Canal area			4/0
71A	140'	north preaccess normal AFU area			2/0
71B	140'	South preaccess normal AFU area			2/0

TABLE 3.3-11 (Continued) Page # 7

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		MAIN STEAM SUPPORT STRUCTURE			
72	80'	Turbine Driven Aux. Feedpump Rm.	-	-	0/3
73	80'	Motor Driven Aux. Feedpump Rd.			1/1
74A	100', 120' + 140'	Main Steam Isol + Dump valve Area - North			~~6/0~~ 0/6
74B	100', 120' + 140'	Main Steam Isol + Dump valve Area - South			~~6/0~~ 0/6
		Outside Areas			
83		Condensate Storage Tank Pump House			2/0
84A		Spray Pond Pump House - TR. A			1/0
84B		Spray Pond Pump House - TR. B			1/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS		
			HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)

(x/y): x is the number of early warning fire detection and notification only instruments.

y is the number of actuation of fire suppression systems and early warning fire detection and notification instruments.

* The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. Three 50% capacity fire suppression pumps, each with a capacity of 1500 gpm ±10% with their discharge aligned to the fire suppression header,

b. Two separate water supply tanks, each with a minimum contained volume of 300,000 gallons (23 FEET 1½ INCHES), and

c. An OPERABLE flow path capable of taking suction from the T01-A tank and the T01-B tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.5, and 3.7.11.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position, WHEN REQUIRED TO BE OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each pump develops ~~at least~~ 1500 gpm ±10% at a system head of 125 psig,

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that each fire suppression pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 85 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.11.1.2 The fire pump diesel engines shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by verifying:

1. The diesel fuel oil day storage tanks each contain at least 315 gallons of fuel, and

2. The diesel engines start from ambient conditions and operate for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.1.1.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

   1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems, listed in Table 3.7-3, shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal/smoke test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

PROOF AND REVIEW

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

a. Lower Cable Spreading Room Zone 14 - Control Building 120 ft Elevation

b. Upper Cable Spreading Room Zone 20 - Control Building 160 ft Elevation

c. Diesel Generator Room, one Train A, one Train B Zone 21 - Diesel Generator Building 100 ft Elevation

d. Fuel Oil Day Tank Vaults, one Train A, one Train B Zone 23 - Diesel Generator Building 131 ft Elevation

e. Containment Spray Pumps Room, one Train A, one Train B Zone 30 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

f. High Pressure Safety Injection Pump Rooms, one Train A, one Train B Zone 31 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

g. Low Pressure Safety Injection Pump Rooms, one Train A, one Train B Zone 32 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

h. Electrical Penetration Room, one Train A (Channel C) Zone 42A - Auxiliary Building 100 ft Elevation

i. Electrical Penetration Room, one Train B (Channel B) Zone 42B - Auxiliary Building 100 ft Elevation

j. Corridors Zone 42C - Auxiliary Building 100 ft Elevation

k. Corridors Zone 42D - Auxiliary Building 100 ft Elevation

l. Electrical Penetration Rooms, one Train A (Channel A) Zone 47A - Auxiliary Building 120 ft Elevation

m. Electrical Penetration Rooms, one Train A (Channel D) Zone 47B - Auxiliary Building 120 ft Elevation

n. Central Corridors Zone 52A - Auxiliary Building 120 ft Elevation

o. Central Corridors Zone 52D - Auxiliary Building 120 ft Elevation

p. Turbine-Driven Auxiliary Feed Pump Room Zone 72 - Main Steam Support Structure 81

q. ESF Transformers and 13.8 kV Switchgear - each transformer Zone 76 - Outside Areas 100 ft Elevation

r. Compartments between Auxiliary & Control Buildings between 74 ft & 156 ft 4 inch Elevation on Trains A & B Zone 86

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

1. Lower Cable Spreading Room Zone 14 - Control Building 120 ft Elevation.

   a. System 1.
   b. System 2.
   c. System 3.
   d. System 4.
   e. System 5.
   f. System 6.

2. Upper Cable Spreading Room Zone 20 - Control Building 160 ft Elevation.

3. Diesel Generator Room, Train A, Zone 21 - Diesel Generator Building 100 ft Elevation.

4. Diesel Generator Room, Train B, Zone 21 - Diesel Generator Building 100 ft Elevation.

5. Fuel Oil Day Tank Vault, Train A, Zone 23 - Diesel Generator Building 131 ft Elevation.

6. Fuel Oil Day Tank Vault, Train B, Zone 23 - Diesel Generator Building 131 ft Elevation.

7. Low Pressure Safety Injection Pump Room, Train B, Zone 32 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation.

8. Electrical Penetration Room, Train A (Channel C) Zone 42A - Auxiliary Building 100 ft Elevation.

9. Electrical Penetration Room, Train B (Channel B) Zone 42B - Auxiliary Building 100 ft Elevation.

10. Charging Pumps A, B and E Zones 46. Corridors Zone 42C Auxiliary Building 100 ft Elevation.

11. Corridors Zone 42D - Auxiliary Building 100 ft Elevation.

12. Electrical Penetration Room, Train A (Channel A) Zone 47A - Auxiliary Building 120 ft Elevation.

13. Electrical Penetration Room, Train A (Channel D) Zone 47B - Auxiliary Building 120 ft Elevation.

14. Central Corridors Zone 52A - Auxiliary Building 120 ft Elevation.

15. Central Corridors Zone 52D - Auxiliary Building 120 ft Elevation.

16. Turbine - Driven Auxiliary Feed Pump Room Zone 72 - Main Steam Support Structure 81.

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

17. Compartments between Auxiliary & Control Buildings between 74 ft & 156 ft 4 inch Elevation on Train A, Zone 86.

18. Compartments between Auxiliary & Control Buildings 74 ft & 156 ft. 4 inch Elevation on Train B, Zone 86.

19. Main Steam Support Structure 100 ft through 140 ft Elevation.

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following low pressure CO_2 systems shall be OPERABLE.

a. ESF Switchgear Room; one Train A, one Train B Zone 5 Control Building 100 ft Elevation

b. Battery Rooms; one Train A (Channel C) one Train B (Channel D) Zone 8 Control Building 100 ft Elevation

c. Battery Rooms; one Train A (Channel A) one Train B (Channel B) Zone 9 Control Building 100 ft Elevation

APPLICABILITY: Whenever equipment protected by the CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.11.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank weight to be greater than 10000 lb and pressure to be greater than 275 psig, and

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying:

1. The system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.5 The fire hose stations shown in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE, except that fire hose stations located in containment shall have their containment isolation valves closed in MODES 1, 2, 3, 4*, and 5*.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-4 inoperable, route a fire hose to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself,

   a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose and related valves shall be mounted above the hose and at the inoperable hose station.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.
   2. Removing the hose for inspection and reracking, and
   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

*If maintenance is to be performed in containment during MODE 4 or 5, the fire hose stations located in containment shall have their containment isolation valves open during the period the maintenance is being performed.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

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TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK IDENTIFICATION
Containment SW	80'	HS #03
Containment NW	80'	HS #04
Containment NE	100'	HS #05
Containment SE	100'	HS #06
Containment SW	100'	HS #07
Containment NW	100'	HS #08
Containment NE	120'	HS #09
Containment SE	120'	HS #10
Containment SW	120'	HS #11
Containment NW	120'	HS #12
Auxiliary Bldg. North Corridor - W	40'	HS #17
Auxiliary Bldg. North Corridor - E	40'	HS #18
Auxiliary Bldg. North Corridor - W	51'6"	HS #21
Auxiliary Bldg. North Corridor - E	51'6"	HS #22
Auxiliary Bldg. SE	70'	HS #23
Auxiliary Bldg. SW	70'	HS #24
Auxiliary Bldg. NW	70'	HS #25
Auxiliary Bldg. North Center Corridor	70'	HS #26
Auxiliary Bldg. NE	70'	HS #27
Auxiliary Bldg. NW	88'	HS #30
Auxiliary Bldg. NE	88'	HS #31
Auxiliary Bldg. SW	100'	HS #33
Auxiliary Bldg. East Corridor	120'	HS #37
Auxiliary Bldg. SW	120'	HS #38
Control Bldg. SW	74'	HS #86
Control Bldg. E	74'	HS #87
Control Bldg. SW	100'	HS #88
Control Bldg. East by Elevator	100'	HS #89
Control Bldg. SW	120'	HS #90
Control Bldg. SW	140'	HS #92
Control Bldg. SW	160'	HS #94
Control Bldg. SE	100'	HS #108
Fuel Bldg. South	100'	HS #97
CONTAINMENT NE	80	HS 1
CONTAINMENT SE	80	HS 2
CONTAINMENT NE	140	HS 13
CONTAINMENT SW	140	HS 14

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RECEIVED AUG 1984 NUCLEAR OPERATIONS

PLANT SYSTEMS

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2-1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged. For Damage

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

3. Performing a flow check of each hydrant to verify its OPERABILITY.

3/4.7.12 FIRE-RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire-rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: When the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire-rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire-rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

A. That each locked-closed fire door is closed at least once per 7 days.

B. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, ~~and~~ perform~~ing~~ a functional test at least once per 18 months.

c.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION ACTUATED

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are estimated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

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3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

ACTUATED

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Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

The FPER (Fire Protection Evaluation Report) fire zones, listed in Table 3.3-11, Fire Detection Instruments, are discussed in Section 9B of the PVNGS FSAR.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected area(s) until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. ~~Level measurements are made by either a U.L. or F.M. approved method.~~ CO2

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.12 FIRE-RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.13 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

SEP 26 1984

MEMORANDUM FOR: Thomas M. Novak, Assistant Director for Licensing, DL

FROM: R. Wayne Houston, Assistant Director for Reactor Safety, DSI

SUBJECT: PALO VERDE UNIT 1 - RSB REVIEW OF TECHNICAL SPECIFICATIONS

Plant Name:	Palo Verde, Unit 1
Docket No.:	50-528
Licensing Stage:	OL
Responsible Branch:	Licensing Branch #3
Project Manager:	E. Licitra
Review Branch:	Reactor Systems Branch
Review Status:	Awaiting Information

The Reactor Systems Branch (RSB) has completed its review of the draft Palo Verde 1 Technical Specifications dated August 13, 1984. The RSB has reviewed only those sections that are within its scope of responsibility.

On Thursday, September 20, 1984, members of RSB met with CE and APS representatives concerning the Palo Verde Unit 1 Technical Specifications. All RSB concerns were discussed during that meeting. As a result of this meeting, some concerns were deleted, while some others were recategorized as requiring further staff evaluation.

RSB's concerns fall into two categories: (1) Items that should be resolved as soon as possible (these are described in Enclosure 1 to this memorandum); and (2) items that require further staff evaluation (these items will be the subject of a forthcoming memorandum from B. Sheron to C. Thomas).

CONTACT: C. Liang, x24754

Reactor Systems Branch believes that all items addressed in Enclosure 1 to this memo need to be resolved prior to issuance of the license. We are continuing to work with the applicant on resolution of these issues.

Original Signed By
R. Wayne Houston

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosure: As stated

cc: R. Bernero
C. Thomas
E. Licitra
D. Brinkman
G. Knighton
J. Wermiel
J. Donohew
B. Buckley

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Palo Verde Unit 1

Request for Additional Information

on Technical Specifications

(RSB)

1. Reactor Protective Instrumentation Setpoints (Table 2.2-1, Section 2.2, page 2-3 and 2-4)

a) Provide basis for the trip setpoint of the high pressurizer pressure on the Supplementary Protection System (SPS).

b) Table 15.0-4 of the FSAR indicates that the analysis setpoint of the high pressurizer pressure is 2450 psia. Explain how the SPS pressurizer pressure-high with an allowable setpoint value of $\leq$ 2439 psia plus instrument uncertainty could ensure the plant operation within the conditions covered by the safety analysis.

c) Confirm that the overpower setpoint in Table 15.0-4 of FSAR will be modified to 117%.

d) Provide basis of the variable overpower allowable setpoint value of 11.0%/min in light of the safety analysis assumptions.

2. Reactor Coolant System Process Variable LCOs

Are the values used for process variable LCOs indicated values from the instrumentation or the actual values in the systems? If they are actual values, please explain how instrument uncertainly is accounted for when determining if an LCO is met or exceeded.

3. Moderator Temperature Coefficient (Section 3.1.1.3, page 3/4 1-4)

The Technical Specification (3.1.1.3 and Figure 3.1-1) permit plant operation at Modes 1 and 2 with a moderator temperature coefficient range of between 0.22×10^{-4} and -3.5×10^{-4}. The single reactor coolant pump rotor seizure with loss of offsite power event was analyzed at full power with a moderator coefficient of 0.0. Would the event analyzed at full power with a moderator coefficient of 0.0 be more limiting than operating at lower power level with a moderator coefficient of a value specified in Figure 3.1-1.

4. Boron Injection Flow Paths (Section 4.1.2.2.b, page 3/4 1-8)

Provide basis for the minimum flow of 26 gpm to the RCS from the boron injection flow path specified in the surveillance requirements.

5. Boron Dilution (Section 3.1.2.7, page 3/4 1-16 through 3/4 1-16d)

Provide basis for the monitoring frequencies for boron dilution detection listed in Tables 3.1-1 through 3.1-5.

6. RPS/ESF Response Times (Table 3.3-2, page 3/4 3-9 and 3.3-5, page 3/4 3-24 through 3/4 3-26)

Provide the basis for RPS/ESF response times listed in these tables or refer to the assumptions made in chapter 15 of FSAR. Also explain why main feedwater isolation response time has not been included in Table 3.3-5.

7. Overpressure Protection System (Section 3.4.8.3, page 3/4 4-32)

Figure 3/4 3.3-1 should be modified to add a curve of Pressure/Temperature limits for RCS cooldown at a rate of 40°F/hour which is used as the basis of the LCO in Section 3.4.8.1

8. <u>Steam Generator Water Level</u> (Section 3/4.4)

Explain why there is no LCO on the steam generator water level. What assurance is there that the steam generator water level will not exceed the values assumed in the safety analyses?

9. <u>Operability of the Steam Generators</u> (Section 4.4.1.2.3 and 4.4.1.3.2, page 3/4 4-2 and 3/4 4-4)

These surveillance requirements state that the required steam generator(s) shall be determined operable by verifying the secondary side water level to be 25% of wide range indication at least once per 12 hours. Provide the basis for the 25% steam generator water level.

10. <u>Auxiliary Feedwater System</u> (Section 3.7.1.2, page 3/4 7-4)

Section 4.7.1.2 should be modified to include surveillance test of each AFW pump to verify the required pump head and flow rate.

11. <u>Auxiliary Pressurizer Spray System</u> (Section 3/4.4)

The current Palo Verde technical specifications do not include a section to address limiting conditions for operation and surveillance requirements on the auxiliary pressurizer spray system (APSS). It is the staff's understanding that the APSS is required for RCS depressurization during plant shutdown per the requirement of the BTP RSB 5-1 (i.e., plant cooldown using only safety-related equipment) and during post-SGTR operation. Does the applicant intend to develop appropriate technical specifications for the APSS. If not, provide the technical basis for not doing so.

12. Cold Shutdown With Loops Filled (Section 3.4.1.4.1, page 3/4 4-5)

The limiting condition for operation specified in this section will permit the plant to operate in Mode 5 with the reactor coolant loops filled, only one SDCS loop in operation, plus two steam generators having 25% water level. Explain how the plant could be maintained in Mode 5 assuming a failure of the operating SDCS loop. Verify that sufficient natural circulation could be achieved during Mode 5.

13. Safety Valves (Section 3.4.2.1, page 3/4 4-7)

Section 3.4.2.1.b permits that the provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in Mode 4. Provide the basis for this technical specification provision.

14. Pressure/Temperature Limits (Section 3.4.8.1, page 3/4 4-28)

Verify and modify the temperature limits indicated in this section consistent with Figure 3/4 3.3-2.

15. Reactor Coolant System Vents (Section 3.4)

The current Palo Verde technical specifications do not include a section to address limiting conditions for operation and surveillance requirements on the reactor coolant system vents. It is the staff's understanding that the applicant takes credit for RCS vents to depressurize the RCS during shutdown per BTP RSB 5-1. Does the applicant intend to develop appropriate technical specification for the RCS vents. If not, provide the technical basis for not doing so.

16. Atmospheric Steam Dump Values (Section 3/4.7)

The current Palo Verde technical specifications do not include a section to address limiting conditions for operation and surveillance requirements on the atmospheric steam dump valves (ADVs).

Since the ADVs are required during initial phase of plant shutdown per the requirements of the BTP RSB 5-1 (i.e., plant cooldown using only safety-related equipment), and we understand your FSAR Chapter 15 steam generator tube rupture analysis takes credit for these components, explain what assurances exist in the plant that these components will always be operable in accordance with the assumptions made in the safety analyses.

Similarly, the Staff and Commission concluded it was acceptable to defer a decision on the need to install PORVs in your plant based, in part, on the CE PRA study performed for your plant. This PRA placed high reliability on the availability of the ADVs to affect decay heat removal. It is the belief of the staff that the DVs should have technical specifications to assure their operability and availability. If you do not propose technical specifications for the ADVs, then please provide the technical basis for not providing technical specifications, and address how the assurances you are providing are consistent with the reliability assumptions made in your PRA.

17. Safety Injection Tanks (Section 3.5.1, page 3/4 5-1)

Section 3/4.5.1 describes the modes of operation for the safety injection tanks. The basis for this item implies that the values in the technical specification were chosen for compliance with the

accident analyses. Address why there are no specifications for the coolant temperature in SIT. Otherwise, justify why the SIT coolant temperature assumed in the ECCS analyses bounds the maximum temperature the SIT could attain.

18. Special Test Exceptions, Reactor Coolant Loops (Section 3/4.10.3 page 3/4 10-3)
This technical specification permits plant operation up to 5% thermal power on fission heat without any reactor coolant pumps operating for startup or physics test. What safety analyses have been conducted that demonstrate that transients or accidents initiated from this operating condition would be acceptable for Palo Verde Units? Both the steady state and transient reactor coolant system temperature profiles, margin to saturation, core DNBR, and thermal-hydraulic stability should be assessed. The acceptability of the reactor protective system setpoints during various transients and accidents initiated from this condition must also be justified.

19. Safety Valves (Section 3.7.1.1, page 3/4 7-1)
Section 3.7.1.1.b indicates that operation in Mode 3 and 4 may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam line code safety valves associated with the operating steam generator. Describe any safety analysis performed to support the above plant operation.

20. <u>ECCS Surveillance Requirements</u> (Section 4.5.2, page 3/4 5-6)

Provide the basis for flow rates of HPSI and LPSI systems in light of the assumptions used in the safety analysis.

9/14/84...8410020130

PROOF AND REVIEW

DRAFT

TECHNICAL SPECIFICATIONS

PALO VERDE NUCLEAR GENERATING STATION

UNIT NO. 1

DOCKET NO. 50-528

AUG 13 1984

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.

d. Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-D of Specification ~~3.6.3.~~ 3.6.1.1

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 Not Applicable.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

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DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE SYSTEM

1.14 A GASEOUS RADWASTE SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, other than reactor coolant pump controlled bleed-off flow, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.21 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

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PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

PURGE - PURGING

1.24 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.30 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.31 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.36 An UNRESTRICTED AREA shall be any area at or beyond the ~~SITE BOUNDARY~~ STET access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the ~~SITE BOUNDARY~~ used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

STET

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VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.38 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

FPER

1.39 FIRE PROTECTION EVALUATION REPORT AS DISCUSSED IN PVNGS FSAR

PROOF AND REVIEW

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
P	Completed prior to each release.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	COLD LEG TEMPERATURE (T_{cold})
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP #	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350° > T_{cold} > 210°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 210°F$
6. REFUELING**	≤ 0.95	0	$\leq 135°F$

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

See Special Test Exception 3.10.5

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

PROOF AND REVIEW

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.231.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.231, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

PROOF AND REVIEW

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS.

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 2383 psia	≤ 2388 psia
2. Pressurizer Pressure - Low	≥ 1837 psia (2)	≥ 1822 psia (2)
3. Steam Generator Level - Low	≥ 44.2% (4)	≥ 43.7% (4)
4. Steam Generator Level - High	≤ 91.0% (9)	≤ 91.5% (9)
5. Steam Generator Pressure - Low	≥ 919 psia (3)	≥ 912 psia (3)
6. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
7. Reactor Coolant Flow - Low		
a. Rate	≤ 1.05%/s (6)(7)	≤ 1.10%/s (6)(7)
b. Floor	≥ 52.2% (6)(7)	≥ 47.2% (6)(7)
c. Band	≤ 40.0% (6)(7)	≤ ~~42.2%~~ (6)(7) 42.1%
8. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
9. DNBR - Low	≥ 1.231 (5)	≥ 1.231 (5)
B. Excore Neutron Flux		
1. Variable Overpower Trip		
a. Rate	< 10.6%/min of RATED THERMAL POWER (8)	< 11.0%/min of RATED THERMAL POWER (8)
b. Ceiling	< 110.0% of RATED THERMAL POWER (8)	< 111.0% of RATED THERMAL POWER (8)
c. Band	< 9.8% of RATED THERMAL POWER (8)	< 10.0% of RATED THERMAL POWER (8)

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TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	< 0.798% of RATED THERMAL POWER	< 0.895% of RATED THERMAL POWER
b. Shutdown	< 0.798% of RATED THERMAL POWER	< 0.895% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	$\leq$ 2434 psia	$\leq$ 2439 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

PROOF AND REVIEW

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(6) RATE is the maximum rate of decrease of the trip setpoint.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor. SET POINTS ARE % OF 100% POWER FLOW CONDITIONS

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

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TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$ 1.15
61	FC2	Core coolant mass flow rate calibration constant	$\leq$ 0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	$\geq$ 1.02
64	TPC	Thermal power calibration constant	$\geq$ 0.90
65	KCAL	Neutron flux power calibration constant	$\geq$ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

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2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to 1.231 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before ~~emergency~~ AUXILIARY feedwater is required TO PREVENT DEGRADED CORE COOLING.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

PROOF AND REVIEW

Local Power Density - High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of ~~1785~~ 1861 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core

PROOF AND REVIEW

DNBR - Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	≥ 470°F
b.	RCS Cold Leg Temperature-High	≤ 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	≥ ~~1860~~ psia 1861
f.	Pressurizer Pressure-High	≤ ~~2339~~ psia 2388
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 4.28
i.	Quality Margin-Low	≥ 0

Steam Generator Level - High

~~The Steam Generator Level - High trip provides protection in the event of excess feedwater flow. The setpoint for the trip is identical to the main steam isolation setpoint.~~

Reactor Coolant Flow - Low

A FOUR PUMP FLOW COASTDOWN DURING A STEAMLINE BREAK WITH A LOSS OF OFFSITE POWER

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions.

THE STEAM GENERATOR LEVEL-HIGH TRIP IS PROVIDED TO PROTECT THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. SINCE THE TURBINE IS AUTOMATICALLY TRIPPED WHEN THE REACTOR IS TRIPPED, THIS TRIP PROVIDES A RELIABLE MEANS FOR PROVIDING PROTECTION TO THE TURBINE FROM EXCESSIVE MOISTURE CARRY OVER. THIS TRIP'S SETPOINT DOES NOT CORRESPOND TO A SAFETY LIMIT AND NO CREDIT WAS TAKEN IN THE ACCIDENT ANALYSES FOR OPERATION OF THIS TRIP. ITS FUNCTIONAL CAPABILITY AT THE SPECIFIED TRIP SETTING ENHANCES THE OVERALL RELIABILITY OF THE REACTOR PROTECTION SYSTEM

PROOF AND REVIEW

Pressurizer Pressure - High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications [~~such as calorimetric measurements for~~] OF power level, ~~and~~ RCS flow rate, [~~and incore detector signals for~~] axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

PROOF AND REVIEW

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

PROOF AND REVIEW

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unles the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual specifications.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with 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 specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

PROOF AND REVIEW

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{cold} GREATER THAN 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION: A) WITH KEFF ≥ 1, COMPLY WITH SPECIFICATION 3.1.3.6

b) With the SHUTDOWN MARGIN less than 6.0% delta k/k AND KEFF < 1, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 6.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

PROOF AND REVIEW

SHUTDOWN MARGIN - T_{cold} LESS THAN OR EQUAL TO 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 K_{eff} shall be determined to be equal to or less than 0.98 at least once per 24 hours, when the RCS water level is drained below the pressurizer low level instrument tap, by performing a reactivity balance considering the factors listed in Specification 4.1.1.2.1b.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

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*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

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MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{cold}) shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{cold}) less than 552°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{cold}) shall be determined to be greater than or equal to 552°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{cold} is less than 557°F.

#With K_{eff} greater than or equal to 1.0.

*See Special Test Exception 3.10.5.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. If only the spent fuel pool in Specification 3.1.2.5a. is OPERABLE, a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the Reactor Coolant System.

b. If only the refueling water tank in Specification 3.1.2.5b. is OPERABLE, a flow path from the refueling water tank via either a charging pump, a high pressure safety injection pump, or a low pressure safety injection pump to the Reactor Coolant System.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following ~~three~~ FOUR boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

D. A FLOW PATH FROM THE REFUELING WATER TANK VIA A HIGH PRESSURE SAFETY INJECTION PUMP TO THE REACTOR COOLANT System

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm to the Reactor Coolant System.

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CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump* or one high pressure safety injection pump or one low pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump or low pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Whenever the reactor coolant level is below the bottom of the pressurizer in MODE 5, one and only one charging pump shall be OPERABLE, by verifying at least once per every 7 days that power is removed from the remaining charging pumps.

PROOF AND REVIEW

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

PROOF AND REVIEW

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F; restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4...2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

PROOF AND REVIEW

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

   1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Table 3.1-1 by either boronometer or RCS sampling.*

b. With both startup channel high neutron flux alarms inoperable:

   1. Determine the RCS boron concentration by both boronometer and RCS sampling when entering MODE 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 - 3.1-5, as applicable, by both boronometer and RCS sampling. If one of the methods of determining the RCS boron concentration is not available, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one additional method for detecting a boron dilution is restored to OPERABLE status.

      See attached attachment 3/4 1-14A

   2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK:

   1. At least once per 12 hours.

   2. When initially setting setpoints at the following times:

      a) One hour after a reactor trip.

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

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ATTACHMENT 3/4 1-14 A

With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by either boronmeter and RCS sampling* or by independent collection and analysis of two RCS samples when entering Mode 3,4,or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable montioring frequency in Tables 3.1-1 through 3.1.5, as applicable, by either boronmeter and RCS sampling* or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the methods for determining and confirming RCS boron concentration is restored.

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SURVEILLANCE REQUIREMENTS (Continued)

b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

* WITH ONE OR MORE REACTOR COOLANT PUMPS (RCP) OPERATING, THE SAMPLE SHOULD BE OBTAINED FROM THE HOT LEG. WITH NO RCP OPERATING, THE SAMPLE SHOULD BE OBTAINED FROM THE DISCHARGE LINE OF THE LOW PRESSURE SAFETY INJECTION (LPSI) PUMP OPERATING IN THE SHUTDOWN COOLING MODE.

PROOF AND REVIEW

TABLE 3.1-1

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} > 0.98$

OPERATIONAL MODE	Number of OPERABLE Charging Pumps 0	1	2	3
3	12 hours	1 hour	Operation not allowed	
4	12 hours	1 hour	Operation not allowed	
5 RCS filled	8 hours	1 hour	Operation not allowed	
5 RCS partially drained	Operation not allowed			
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.98 \geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of OPERABLE Charging Pumps 0	1	2	3
3	12 hours	2.5 hours	1 hour	0.5 hours
4	12 hours	2.5 hours	1 hour	0.5 hours
5 RCS filled	8 hours	2.5 hours	1 hour	0.5 hours
5 RCS partially drained	8 hours	0.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of OPERABLE Charging Pumps			
	0	1	2	3
3	12 hours	3.5 hours	1.5 hours	1 hour
4	12 hours	3.5 hours	1.5 hours	1 hour
5 RCS filled	8 hours	3.5 hours	1.5 hours	1 hour
5 RCS partially drained	8 hours	1 hour	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of OPERABLE Charging Pumps 0	1	2	3
3	12 hours	5 hours	2 hours	1 hour
4	12 hours	5 hours	2 hours	1 hour
5 RCS filled	8 hours	5 hours	2 hours	1 hour
5 RCS partially drained	8 hours	1.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

TABLE 3.1-5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERABLE CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of OPERABLE Charging Pumps 0	1	2	3
3	12 hours	6 hours	3 hours	1.5 hours
4	12 hours	6 hours	3 hours	1.5 hours
5 RCS filled	8 hours	6 hours	3 hours	1.5 hours
5 RCS partially drained	8 hours	2 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

PROOF AND REVIEW

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. IN ADDITION, THE POSITION OF THE PART LENGTH CEAS GROUPS SHALL BE LIMITED TO THE INSERTION LIMITS SHOWN ON FIGURE 3.1-2A

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full-length or part-length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

C ~~d~~. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches ~~but less than or equal to 19 inches~~, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-2B and that within 1 hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

   a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

D ~~e~~. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits (Figure 3.1-3) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

~~f~~. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, ensure that the CEA is: (1) within its above specified alignment requirements and, (2) either fully withdrawn or, if in full-length CEA group 5, within the Long Term Steady State Insertion Limits of Figure 3.1-3. Then operation in MODES 1 and 2 may continue.

E ~~g~~. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

DATE	SUBJECT	
DATE		JOB NO.

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PROOF AND REVIEW

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.*

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

PROOF AND REVIEW

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating, or part-length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

PROOF AND REVIEW

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 4 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{cold} greater than or equal to 552°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

PROOF AND REVIEW

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SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 144.75 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 144.75 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Withdraw the CEA to at least 144.75 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 144.75 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

#With K_{eff} greater than or equal to 1.

PROOF AND REVIEW

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, and to the insertion limits## shown on Figure 3.1-3** when the COLSS is in service or shown on Figure 3.1-4** when the COLSS is not in service. The CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Months.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using ~~the above~~ figures 3.1-3 OR 3.1-4.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-3 or Figure 3.1-4 are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.

**CEAs are fully withdrawn in accordance with Figure 3.1-3 or Figure 3.1-4 when withdrawn to at least 144.75 inches.

##Following a reactor power cutback in which (1) Regulating Group~~s~~ ~~4 and/or~~ 5 OR REG. GROUPS 4 AND 5 are dropped or (2) Regulating Groups ~~4 and/or~~ 5 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) sequentially inserted, OR REG GROUPS 4 AND 5 the Transient Insertion Limit of Figure 3.1-3 OR FIGURE 3.1-4 can be exceeded for up to 2 hours.

PROOF AND REVIEW

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ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

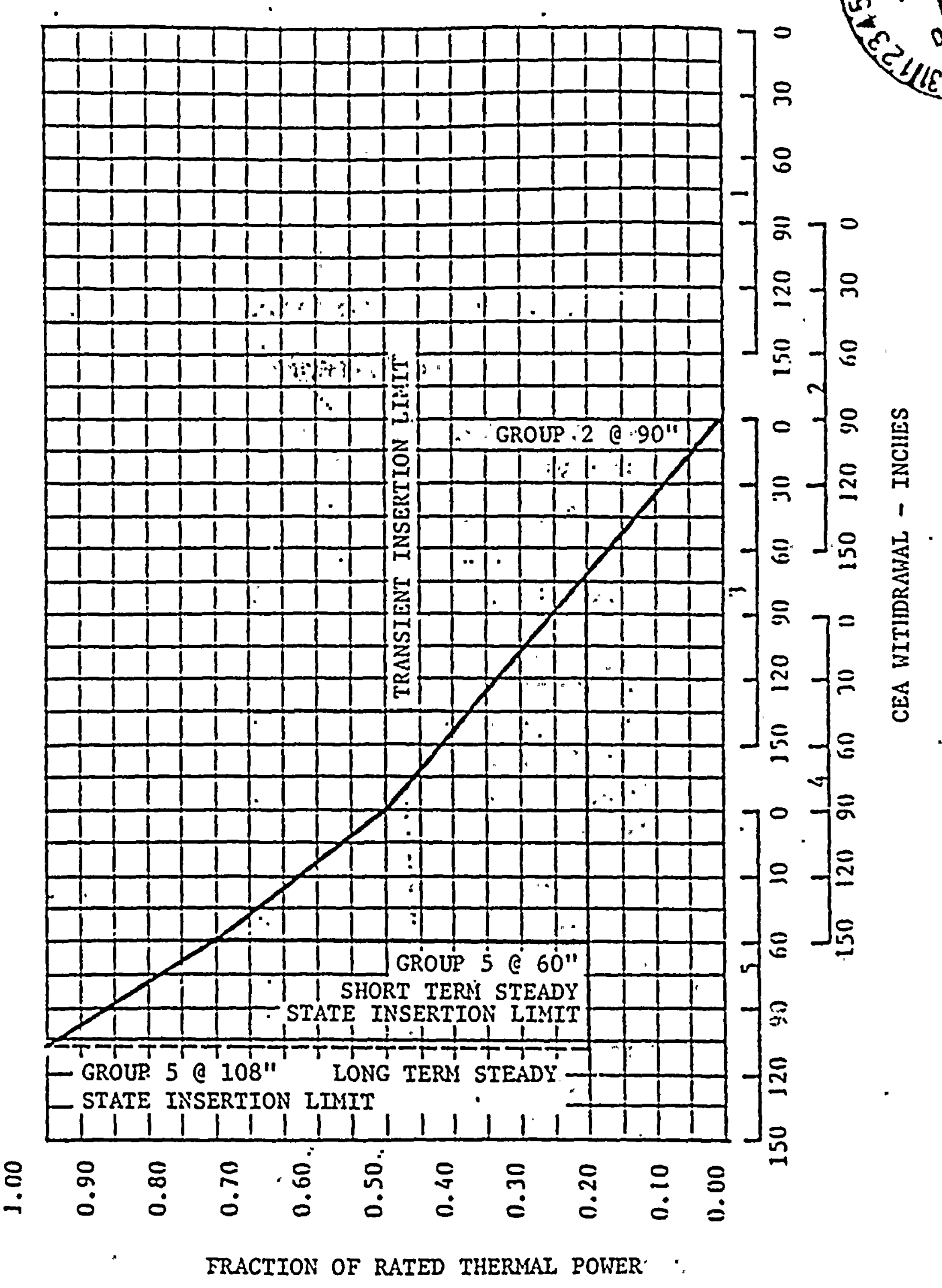

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS IN SERVICE)

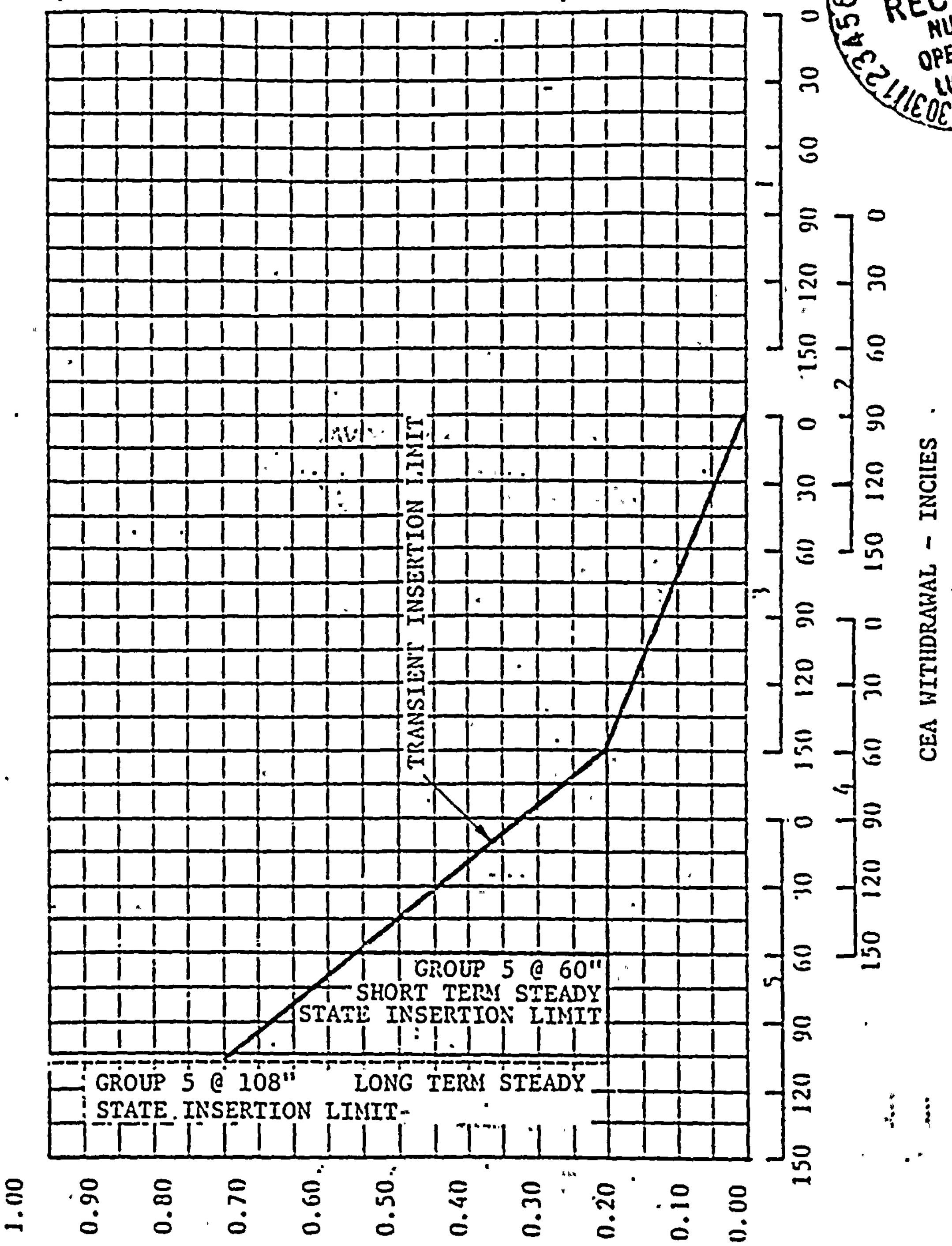

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FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS OUT OF SERVICE)

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 14.0 kW/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kW/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is less than or equal to 14.0 kW/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on 14.0 kW/ft.

PROOF AND REVIEW

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With an F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

*See Special Test Exception 3.10.2.

PROOF AND REVIEW

3/4.2.3 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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*See Special Test Exception 3.10.2.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT less than or equal to the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

PROOF AND REVIEW

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POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable, or in accordance with the requirements of Action 6c of Table 3.3-1. ~~when COLSS and both CEACs are inoperable~~.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 ~~days:~~ EFPD.

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)*
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

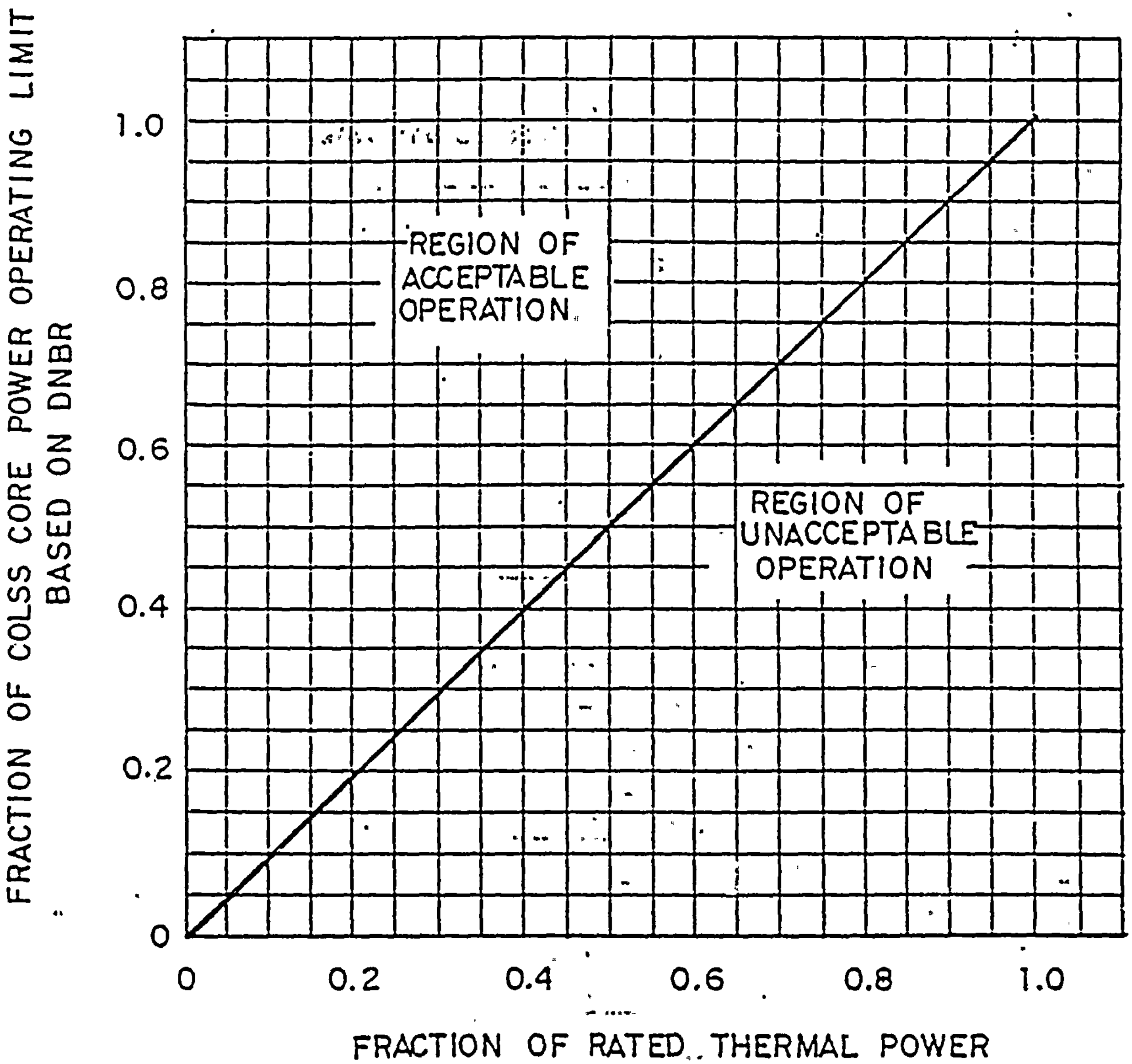

FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS
(COLSS IN SERVICE)

DATE	SUBJECT	
DATE		JOB ...

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PROOF AND REVIEW

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 164.0 x 10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than its limit at least once per 12 hours.

PROOF AND REVIEW

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.*

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.

DATE	SUBJECT	
DATE		JOB NO.

THIS GRAPH WILL BE COMING IN

ABOUT A WEEK.

PROOF AND REVIEW

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq$ ASI ≤ 0.28

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq$ ASI $\leq + 0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

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3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restart periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	$2^{\#}$, $3^{\#}$
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	$2^{\#}$, $3^{\#}$
6. Containment Pressure - High	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	$2^{\#}$, $3^{\#}$
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
9. DNBR - Low	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	$2^{\#}$, $3^{\#}$
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	$2^{\#}$, $3^{\#}$
	4	2	3	3*, 4*, 5*	8
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1, 2	$2^{\#}$, $3^{\#}$, 7

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
D. Supplementary Protection System					
Pressurizer Pressure - High	4 (f)	2	4	1, 2	8
II. RPS LOGIC					
A. Matrix Logic	6	1	3	1, 2	1
	6	1	3	3*, 4*, 5*	8
B. Initiation Logic	4	2	4	1, 2	5
	4	2	4	3*, 4*, 5*	8
III. RPS ACTUATION DEVICES					
A. Reactor Trip Breaker	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8
B. Manual Trip	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6l. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER-OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1. The trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. AFTER 7 DAYS, OPERATION MAY CONTINUE PROVIDED THAT THE CONDITIONS OF ACTION ITEM 6.b OR 6.c ARE MET

b. With both CEACs inoperable and COLSS in ~~operation~~, SERVICE operation may continue provided that:

1. Within 1 hour:

~~a)~~ The margins required by Specification 3.2.4 are increased and maintained at a value equivalent to or greater than the percentage of RATED THERMAL POWER shown on Figure 3.3-1.

b) The Reactor Power Cutback System is ~~disabled~~. (PLACED OUT OF SERVICE

A) OPERATION IS RESTRICTED TO THE LIMITS SHOWN IN Fig 3.3-1. THE DNBR MARGIN REQUIRED BY SPECIFICATION 3.2.4 IS REPLACED BY THIS RESTRICTION WHEN BOTH CEAC'S ARE INOPERABLE AND COLSS IS IN OPERATION

b) THE LINEAR HEAT RATE MARGIN REQUIRED BY SPECIFICATION 3.2.1 IS MAINTAINED

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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2. Within 4 hours:

a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status.~~ BE INDICATE THAT BOTH CEACS ARE INOPERABLE

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above; then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. FOLLOWING A CEA MISALIGNMENT, WITH BOTH CEAC'S INOPERABLE AND COLSS IN OPERATION, OPERATION MAY CONTINUE PROVIDED THAT:

WITHIN 1 HOUR:

A) THE POWER IS REDUCED TO 85% OF THE PRE-MISALIGNED POWER BUT NEED NOT BE REDUCED TO LESS THAN 60% OF RATED POWER.

b) REFER TO SECTION 3.1.3; MOVABLE CONTROL ASSEMBLIES, FOR FURTHER SPECIFICATIONS ON CEA MISALIGNMENT.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

1. Within 1 hour:

a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by ~~1.18 and the~~ 1.19 resulting value is re-entered into the CPCs.

b) The Reactor Power Cutback System is ~~disabled~~. PLACED OUT OF SERVICE

c) THE COLSS OUT OF SERVICE LIMIT LINE, SECTION 3.2.4 IS NOT APPLICABLE TO THIS MODE OF OPERATION.

PROOF AND REVIEW

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to ~~the inoperable status~~. BE INDICATED THAT BOTH CEAC'S ARE INOPERABLE

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. FOLLOWING A CEA MISALIGNMENT, WITH BOTH CEAC'S AND COLSS INOPERABLE, OPERATION MAY CONTINUE PROVIDED THAT: WITHIN 1 HOUR:

   a) THE POWER IS REDUCED TO 85% OF THE PRE-MISALIGNED POWER, BUT NEED NOT BE REDUCED TO LESS THAN 50% OF RATED POWER.

   b) REFER TO SECTION 3.1.3, MOVABLE CONTROL ASSEMBLIES, FOR FURTHER SPECIFICATIONS ON CEA MISALIGNMENT

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - Wi‥ the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breakers within the next hour.

DATE	SUBJECT	
DATE		JOB NO.

THIS GRAPH WILL BE COMING IN

ABOUT A WEEK.

DATE	SUBJECT	
DATE		JOB NO

THIS GRAPH WILL BE COMING IN
ABOUT A WEEK.

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TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME	
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 1.15 seconds	
2. Pressurizer Pressure - Low	≤ 1.15 seconds	
3. Steam Generator Level - Low	≤ 1.15 seconds	
4. Steam Generator Level - High	≤ 1.15 seconds	
5. Steam Generator Pressure - Low	≤ 1.15 seconds	
6. Containment Pressure - High	≤ 1.15 seconds	
7. Reactor Coolant Flow - Low	≤ ~~0.75~~ second	0.65
8. Local Power Density - High		
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second*	0.75
b. CEA Positions	≤ ~~0.22~~ second**	1.35
c. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second**	0.75
9. DNBR - Low		
a. Neutron Flux Power from Excore Neutron Detectors	≤ ~~0.61~~ second*	0.75
b. CEA Positions	≤ ~~0.22~~ second**	1.35
c. Cold Leg Temperature	≤ ~~0.81~~ second##	0.75
d. Hot Leg Temperature	≤ ~~0.81~~ second##	0.75
e. Primary Coolant Pump Shaft Speed	≤ ~~0.52~~ second#	0.75
f. Reactor Coolant Pressure from Pressurizer	≤ ~~0.48~~ second###	0.75
g. CEA Positions: CEAC Penalty Factor	≤ ~~0.41~~ second**	0.75
B. Excore Neutron Flux		
1. Variable Overpower Trip	≤ 1.15 second*	
2. Logarithmic Power Level - High		
a. Startup and Operating	≤ 0.55 second*	
b. Shutdown	≤ 0.55 second*	

PROOF AND REVIEW

PALO VERDE - UNIT 1

TABLE 3[illegible] (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

[#Response time shall be measured from the onset of a two-out-of-four reactor coolant pump coastdown.]

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 13 seconds. Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accomodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

THE PULSE TRANSMITTERS MEASURING PUMP SPEED ARE EXEMPT FROM RESPONSE TIME TESTING. THE RESPONSE TIME SHALL BE MEASURED FROM THE PULSE SHAPE INPUT

PROOF AND REVIEW

TABLE 3.3-2a

INCREASES IN BERRO, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME (τ)	BERRO INCREASE (%)	BERR2 INCREASE (%)	BERR4 INCREASE (%)
$\tau \leq$ 8.0 sec 0	0	0	0
8.0 sec < $\tau \leq$ 10.0 sec	2.5	2.0	1.0
10.0 sec < $\tau \leq$ 13.0 sec	6.0	4.0	6.0

NOTE: BERR term increases are not cumulative. For example, if the time constant changes from the range of 8.0 < $\tau \leq$ 10.0 sec to the range 10.0 < $\tau \leq$ 13.0, the BERRO increase from its original ($\tau \leq$ 8.0 sec) value is 6.0 not 2.5 + 6.0.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	D (2, 4), R (4, 5)	M, R (6)	1, 2
9. DNBR - Low	S	D (2, 4), R (4, 5) M (8), S (7)	M, R (6)	1, 2
B. Excore Neutron Flux				
1. Variable Overpower Trip	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5 and *
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M, S/U (1) R	1, 2, 3*, 4*, 5*

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

Either using the Reactor Coolant Pump Differential Pressure Instrumentation and the ultrasonic flow meter adjusted pump curves (conservatively compensated for measurement uncertainties) or

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

PROOF AND REVIEW

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. SAFETY INJECTION (SIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a), 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
II. CONTAINMENT ISOLATION (CIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
II. CONTAINMENT ISOLATION (Continued)					
3. Manual CIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
4. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
III. CONTAINMENT SPRAY (CSAS)					
A. Sensor/Trip Units					
Containment Pressure -- High - High	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual CSAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
IV. MAIN STEAM LINE ISOLATION (MSIS)					
A. Sensor/Trip Units					
1. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(b), 4	13*, 14*
2. Steam Generator Level - High	4/steam generator	2/steam generator	3/steam generator	1, 2, 3, 4	13*, 14*
3. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual MSIS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.[illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
V. RECIRCULATION (RAS)					
A. Sensor/Trip Units					
Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual RAS	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)					
A. Sensor/Trip Units					
1. Steam Generator #1 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG2 > SG1	4	2	3	1, 2, 3	13*, 14*

TABLE 3 [illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)					
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)					
A. Sensor/Trip Units					
1. Steam Generator #2 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG1 > SG2	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VIII. LOSS OF POWER (LOV)					
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*

PROOF AND REVIEW

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

(a) In MODES 3-6, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-6, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two take twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.61. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2. Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

PROOF AND REVIEW

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17- With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia (1)	≥ 1822 psia (1)
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
2. Pressurizer Pressure - Low	≥ 1837 psia	≥ 1822 psia
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
Containment Pressure High - High	≤ 8.5 psig	≤ 8.9 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	≥ 919 psia	≥ 912 psia
2. Steam Generator Level - High	≤ ~~≥~~ 91.0% NR(2)	≤ ~~≥~~ 91.5% NR(2)
3. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

TABLE 3.3[illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	≥ 8.9% of Span	≥ 8.4% of Span
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	≥ 25.8% WR(3)	≥ 25.3% WR(3)
2. Steam Generator Δ Pressure - SG2 > SG1	≤ 185 psid	≤ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	≥ 25.8% WR(4)	≥ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	≤ 185 psid	≤ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3250 volts	≥ 3250 volts
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2930 to 3744 volts with a 35-second maximum time delay	2930 to 3744 volts with a 35-second maximum time delay

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PROOF AND REVIEW

TABLE NOTATION 3.3-4

ATTACHMENT 3-23-A

1) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

2) % OF THE DISTANCE BETWEEN STEAM GENERATOR UPPER AND LOWER LEVEL NARROW RANGE INSTRUMENT NOZZLES.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

PROOF AND REVIEW

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Containment Purge Valve Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. SRAS	
Containment Sump Recirculation	Not Applicable
f. AFAS	
Auxiliary Feedwater Pumps	Not Applicable

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure - Low	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.3*/11.7**
c. Containment Isolation	≤ 23.1*/13.1**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 16.8*/6.7**
b. Safety Injection (LPSI)	≤ 21.2*/11.7**
c. Containment Isolation	≤ 23.0*/13.0**
d. Main Steam Isolation	≤ 11.0*/11.0**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 31.3*/21.6**
5. Steam Generator Pressure - Low	
a. Main Steam Isolation	≤ 11.1/11.1**
6. Refueling Water Storage Tank - Low	
a. Containment Sump Recirculation	≤ 60.0/60.0**
7. Steam Generator Level - Low	
a. Auxiliary Feedwater (Motor Drive) - SIAS	≤ 26.2*/16.5**
b. Auxiliary Feedwater (Motor Drive) - No SIAS	≤ 26.3*/13.1**
c. Auxiliary Feedwater (turbine drive) - SIAS	≤ 21.1 /21.1**
d. Auxiliary Feedwater (turbine drive) - No SIAS	≤ 21.1 /21.1**
8. Steam Generator Level - High	
a. Main Steam Isolation	≤ 11.0*/11.0**
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator	≤ 23.1*/13.1**

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TABLE 3.3-5 (continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low -	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~916.2~~ 17*/~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~51.2~~ 69*/~~41.2~~ 59**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ ~~916.2~~ 17*/~~96.2~~ 7**
2. Other CIAS actuated valves	≤ ~~51.2~~ 69*/~~41.2~~ 59**
d. Main Steam Isolation	
1. MSIS actuated MSIV's	≤ 6.2*/6.2**
2. MSIS actuated MFIV's	≤ 11.2*/11.2**
3. Other MSIS actuated valves	≤ ~~21.2~~ 69*/~~21.2~~ 59**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 33*/23**

PROOF AND REVIEW

5. Steam Generator Pressure - Low
   a. Main Steam Isolation
      1. MSIS actuated MSIV's — ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's — ≤ 11.2*/11.2**
      3. Other MSIS actuated valves — ≤ ~~21.2~~ 69*/~~21.2~~ 59**

6. Refueling Water Storage Tank - Low
   a. Containment Sump Recirculation — ≤ 45*/45**

7. Steam Generator Level - Low
   a. Auxiliary Feedwater (Motor Drive) - SIAS — ≤ 45*/30**
   b. Auxiliary Feedwater (Motor Drive) - No SIAS — ≤ 45*/30**
   c. Auxiliary Feedwater (Turbine Drive) - SIAS — ≤ 45*/30**
   d. Auxiliary Feedwater (Turbine Drive) - No SIAS — ≤ 45*/30**

8. Steam Generator Level - High
   a. Main Steam Isolation
      1. MSIS actuated MSIV's — ≤ 6.2*/6.2**
      2. MSIS actuated MFIV's — ≤ 11.2*/11.2**
      3. Other MSIS actuated valves — ≤ ~~21.2~~ 69*/~~21.2~~ 59**

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9. Steam Generator ΔP- High
Coincident with Steam Generator
Level Low

a. Auxiliary Feedwater Isolation
from the Ruptured
Steam Generator - $\leq$ ~~31.2*~~ 45 / ~~21.2*~~ 30

PROOF AND REVIEW

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	
Loss of Power	≤ 2.4
11. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
Loss of Power 90% system voltage	≤ 35.0

NOTE: Response time for Motor-Driven and Steam-Driven Auxiliary Feedwater Pumps that start on ESF signals on all ESF signal starts	≤ 60

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
I. SAFETY INJECTION (SIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3, 4
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual SIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
II. CONTAINMENT ISOLATION (CIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CIAS	NA	NA	M	1, 2, 3, 4
4. Manual SIAS	NA	NA	M	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
II. CONTAINMENT ISOLATION (Continued)				
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
III. CONTAINMENT SPRAY (CSAS)				
A. Sensor/Trip Units				
1. Containment Pressure -- High - High	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CSAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4

PROOF AND REVIEW

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
IV. MAIN STEAM LINE ISOLATION (MSIS)				
A. Sensor/Trip Units				
1. Steam Generator Pressure - Low	S	R	M	1, 2, 3, 4
2. Steam Generator Level - High	S	R	M	1, 2, 3, 4
3. Containment Pressure - High	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual MSIS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
V. RECIRCULATION (RAS)				
A. Sensor/Trip Units				
Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual RAS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	R(1) M(1),(2),(3)	1, 2, 3, 4
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)				
A. Sensor/Trip Units				
1. Steam Generator #1 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG2 > SG1	S	R	M	1, 2, 3

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TABLE 4.3[illegible]ntinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG1 > SG2	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	R(1) M(1) (2) (3)	1, 2, 3, 4
VIII. LOSS OF POWER (LOV)				
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	S	R	R	1, 2, 3, 4
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	S	R	R	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

1 ~~(2)~~ Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays ___, ___, ___, ___, ___, ___, and ___ are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

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3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
A. Fuel Pool Area RU-31	1	**	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22 & 24
B. New Fuel Area RU-19	1	*	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22
C. Containment RU-148 RU-149	2	1,2,3,4	≤2 R/hr	1R/hr to 10^{7}R/hr	27
D. Containment Power Access Purge Exhaust RU-37 & RU-38	1	#	≤2.5mR/hr	10^{-1} to 10^{4}mR/hr	25
E. Main Steam					
1) RU-139 A&B	1	1,2,3,4	≤10mR/hr	10^{-3} to 10^{4}R/hr	27
2) RU-140 A&B	1	1,2,3,4	≤10mR/hr	10^{-3} to 10^{4}R/hr	27
2. Process Monitors					
A. Containment Building Atmosphere RU-1	2	1,2,3,4			23 & 27
1) Particulate			≤2.3×10^{-6}µCi/cc Cs-137	10^{-9} to 10^{-4}µCi/cc	
2) Gaseous			≤6.6×10^{-2}µCi/cc Xe-133	10^{-6} to 10^{-1}µCi/cc	
B. Noble Gas Monitors					
1) Control Room Ventilation Intake RU-29 & RU-30	1	All MODES	≤2×10^{-6}µCi/cc	10^{-7} to 10^{-1}µCi/cc	26

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E 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2) Fuel Building Ventilation Exhaust RU-145	1	**	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27 & 24
3) Condenser Vacuum Pump/ Gland Seal Exhaust RU-141	1	1,2,3,4	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27
4) Plant Vent Gaseous RU-143	1	All MODES	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27
5) Waste Gas Decay Tank Discharge RU-12	1	##	Per ODCM	10^{-3} to 10^{2} µCi/cc	***

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
~~***ACTION in accordance with Table 3.3-23 ACTION 35~~.
#When purge is being used.
~~##During waste gas release.~~

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TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.4.5.1).

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.12).

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.9).

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1. Complete the actions of "A" or "B".

   A. Initiate the preplanned alternate method of monitoring the appropriate parameter(s).

   B. For process monitors, place moveable air monitor in-line or take grab sample at least once per 24 hours.

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
A. Fuel Pool Area RU-31	S	R	M	**
B. New Fuel Area RU-19	S	R	M	*
C. Containment Power Access Purge Exhaust RU-37 & RU-38	P#	R	#### P,W##	##
D. Containment RU-148 & RU-149	S	R	M	1,2,3,4
E. Main Steam RU-139 A&B RU-140 A&B	S	R	M	1,2,3,4
2. Process Monitors				
A. Containment Building Atmosphere RU-1				
1) Particulate	S	R	M	1,2,3,4
2) Gaseous	S	R	M	1,2,3,4
B. Noble Gas Monitors				
1) Control Room Ventilation Intake RU-29 & RU-30	S	R	M	All MODES
2) Fuel Building Ventilation Exhaust RU-145	S	R	M	**
3) Condenser Vacuum Pump/Gland Seal Exhaust RU-141	S	R	M	1,2,3,4

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TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4) Plant Vent RU-143	S	R	M	ALL MODES
5) Waste Gas Tanks Discharge RU-12	P	R	P	###

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
#If purge is in service for greater than 12 hours, perform once per 12-hour period.
##When purge system is in operation.
~~###During waste gas release.~~
THE FUNCTIONAL TEST SHOULD CONSIST OF, BUT NOT LIMITED TO A VERIFICATION OF SYSTEM ISOLATION CAPABILITY BY THE INSERTION OF A SIMULATED ALARM CONDITION

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LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use IF 7 DAYS OR MORE HAVE ELAPSED SINCE LAST USE and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

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SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Accelerometers	
a. Tendon Gallery Floor, 55' level	1
b. R.C.P., Upper Support, 129'6" level	1
c. Steam Generator Base, 101'9" level	1
d. Control Building Floor, 74' level	1
e. Auxiliary Building Floor 40' level	1
f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.)	1
2. Peak Reading Accelerograph	
a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level	1
3. Seismic Triggers	
a. Tendon Gallery Floor, 55' level (Setpoint ~~0.010 g)~~ 0.02g)	1
b. Containment Operating Floor, 140' level (Setpoint 0.010 g)	1
4. Digital Cassette Recorders	
a. Control Room Area, 140' level	1
b. Control Room Area, 140' level	1
c. Control Room Area, 140' level	1
d. Control Room Area, 140' level	1
e. Control Room Area, 140' level	1
f. Control Room Area, 140' level	1
5. Seismic Switches	
a. Tendon Gallery Floor, 55' level	1

	Horizontal	Vertical
Setpoint OBE	0.18 g	0.17 g
Setpoint SSE	0.31 g	0.34 g

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	[illegible] TEST
1. Triaxial Accelerometers			
a. Tendon Gallery Floor, 55' level	N.A.	R	SA
b. R.C.P., Upper Support, 129'6" level	N.A.	R	SA
c. Steam Generator Base, 101'9" level	N.A.	R	SA
d. Control Building Floor, 74' level	N.A.	R	SA
e. Auxiliary Building Floor 40' level	N.A.	R	SA
f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.)	N.A.	R	SA
2. Peak Reading Accelerograph			
a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level	N.A.	R	NA
3. Seismic Triggers			
a. Tendon Gallery Floor, 55' level	N.A.	R	SA
b. Containment Operating Floor, 140' level	N.A.	R	SA
4. Digital Cassette Recorders			
a. Control Room Area, 140' level	M	R	SA
b. Control Room Area, 140' level.	M	R	SA
c. Control Room Area, 140' level	M	R	SA
d. Control Room Area, 140' level	M	R	SA
e. Control Room Area, 140' level	M	R	SA
f. Control Room Area, 140' level	M	R	SA
5. Seismic Switches			
a. Tendon Gallery Floor, 55' level	M	R	SA

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. 1Ø to 50 mph,	Nominal Elev. 35 feet	1
b. 1Ø to 50 mph,	Nominal Elev. 200 feet	1
2. WIND DIRECTION		
a. 0°-360°-180°,	Nominal Elev. 35 feet	1
b. 0°-360°-180°,	Nominal Elev. 200 feet	1
3. AIR TEMPERATURE - DELTA T		
a. -6°F to 6°F,	Nominal Elev. 35 feet-200 feet	1

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 35 feet - 200 feet	D	SA

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REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown system monitoring instrumentation channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5. Each remote shutdown system monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION

INSTRUMENTATION	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Log Neutron Power Level	Remote Shutdown Panel	2
2. Reactor Coolant Hot Leg Temperature	Remote Shutdown Panel	1/loop
3. Reactor Coolant Cold Leg Temperature	Remote Shutdown Panel	1/loop*
4. Pressurizer Pressure	Remote Shutdown Panel	1
5. Pressurizer Level	Remote Shutdown Panel	2
6. Steam Generator Pressure	Remote Shutdown Panel	2/steam generator
7. Steam Generator Level	Remote Shutdown Panel	2/steam generator
8. Refueling Water Tank Level	Remote Shutdown Panel	2
9. Charging Line Pressure	Remote Shutdown Panel	1
10. Charging Line Flow	Remote Shutdown Panel	1
11. Shutdown Cooling Heat Exchanger Temperatures	Remote Shutdown Panel	2
12. Shutdown Cooling Flow	Remote Shutdown Panel	2
13. Auxiliary Feedwater Flow Rate	Remote Shutdown Panel	2/steam generator

*Not required until startup following first refueling outage.

TABLE 4.3-6

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Log Neutron Power Level	M	R
2.	Reactor Coolant Hot Leg Temperature (2)	M	R
3.	Reactor Coolant Cold Leg Temperature (2)*	M	R
4.	Pressurizer Pressure	M	R
5.	Pressurizer Level	M	R
6.	Steam Generator Pressure	M	R
7.	Steam Generator Level	M.	R
8.	Refueling Water Tank Level	M	R
9.	Charging Line Pressure	M	R
10.	Charging Line Flow	M	R
11.	Shutdown Cooling Heat Exchanger Temperatures	M	R
12.	Shutdown Cooling Flow	M	R
13.	Auxiliary Feedwater Flow Rate	M	R

*Not required until startup following first refueling outage.

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE post-accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE [illegible]10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	2	1/loop
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	2	1/loop
4. Pressurizer Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator
8. Refueling Water Storage Tank Water Level	2	1
9. Auxiliary Feedwater Flow Rate	2	1
10. Reactor Cooling System Subcooling Margin Monitor	2	1
11. Pressurizer Safety Valve Position Indicator	1/valve	1/valve
12. Containment Water Level (Narrow Range)	2	1
13. Containment Water Level (Wide Range)	2	1

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature -T_{cold} (Wide Range)	M	R
4. Pressurizer Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Wide Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Pressurizer Safety Valve Position Indicator	M	R
12. Containment Water Level (Narrow Range)	M	R
13. Containment Water Level (Wide Range)	M	R

PROOF AND REVIEW

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each ~~fire detection~~ FIRE zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

X

a. With any, but not more than one-half the total in any fire zone Function A fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

X

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

Y

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

SEE New TABLE

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - CONTROL			
1	74'	Essential Chiller Rm. - Train A			24/0
2	74'	Essential Chiller Rm. - Train B			21/0
3	74'	Cable Shaft - Trains A & B			2/0
86	74-156'4"	Deadspace Compartment - Trains A & B	0/2		0/6
4	100'	Cable Shaft - Trains A & B			2/0
5	100'	ESF Switchgear Rooms - Trains A & B			0/20
6	100'	DC Equip. Rms. - Tr. A (Chan. C) Tr. B (Chan. D)			4/0
7	100'	DC Equip. Rms. - Tr. A (Chan. A) Tr. B (Chan. B)			4/0
8	100'	Battery Rms. - Tr. A (Chan. C) Tr. B (Chan. D)	0/4		0/4
9	100'	Battery Rms. - Tr. A (Chan. A) Tr. B (Chan. B)	0/4		0/4
10	100'	Remote Shutdown Rm.			2/0
11	120'	Cable Shafts - Tr. A & B			2/0
14	120'	Lower Cable Spreading Rm.	0/6		0/44
15	140'	Cable Shaft Tr. A & B			2/0
17	140'	Control Rm. - MCB's & Relay Cabinets			97/0
18	160'	Cable Shafts - Tr. A & B			2/0
[illegible]	160'	Upper Cable Spreading Rm.	0/5		0/44

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - DIESEL GENERATOR			
21	100'	Diesel Generator - Tr. A & B	0/6	0/8	
22	100'	Diesel Generator Control Rms. Tr. A & B			2/0
24	115'	Combustion Air Intake Rms. - Tr. A & B			2/0
23	131'	Fuel Oil Day Tanks - Tr. A & B	0/2		
25	131'	Exhaust Silencer Rm. - Tr. A & B		6/0	
		BUILDING - FUEL			
[illegible]8	100'	Spent Fuel Pool Cooling and Cleanup Pump Areas			3/0
		BUILDING - AUXILIARY			
30	51'-6"	Containment Spray Pump Rms. - Tr. A & B			0/4
31	51'-6"	HPSI Pump Rms. - Tr. A & B			0/4
32	51'-6"	LPSI Pump Rms. - Tr. A & B			0/4
34	70'	ECW Pump Rms. - Tr. A & B			4/0
35	70'	Shutdown Cooling Ht. x Chgr. Tr. A & B			8/0
37	70'	Piping Penetration Rm. - Tr. A & B			4/0
37A	70'	Corridors - East & West			18/0
39	88'	Pipeways - Tr. A & B			25/0
[illegible]12A	100'	Elect. Penetration Rm. Tr. A (Chan. C)	0/1		0/25

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
42B	100'	Elect. Penetration Rm. Tr. B (Chan. B)	0/1		0/24
42C	100'	Corridors - East & Southeast	0/1		3/35
42D	100'	Corridor - West	0/1		0/29
46	100'	Charging Pump and Valve Gallery Rms.			0/9
47A	120'	Elect. Penetration Rm. - Tr. A (Chan. A)	0/1		0/28
47B	120'	Elect. Penetration Rm. Tr. B (Chan. D)	0/1		0/24
48	120'	ECW Surge Tanks Corridor - Tr. A & B			0/2
52A	120'	Central Corridor - West	0/1		4/15
52D	120'	Central Corridor - East	0/1		4/17
54	120'	Reactor Trip Switchgear Rm.	1/0		6/0
		BUILDING - CONTAINMENT**			
66	100'	Containment Cable Tray Area - South	1/0		
67	100'	Containment Cable Tray Area - North	1/0		
66	120'	Containment Cable Tray Area - South	1/0		
67	120'	Containment Cable Tray Area - North	1/0		
68	120'	Steam Generator Areas North & South			12/0
69	120'	Containment Cable Tray Area - East	1/0		

See New Table

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
67	140'	Containment Cable Tray Area - North	1/0		
68	140'	Steam Generator Area - North & South			10/0
70	140'	Cavity Cooling Fans			4/0
71	140'	Charcoal Filter Area - North & South			4/0
		MAIN STEAM SUPPORT STRUCTURE			
72	80'	Turbine Driven Aux. Feedpump Rm.			0/3
73	80'	Motor Driven Aux. Feedpump Rm.			1/1
74	100'	Main Steam Isol. & Dump Valve Area			0/4
74	120'	Main Steam Isol. & Dump Valve Area			0/4
74	140'	Main Steam Isol. & Dump Valve Area			0/4

See 3/4 3-56a

* (x/y): x is the number of Function A (early warning fire detection and notification only) instruments. X

y is the number of Function B (actuation of fire suppression systems and early warning and notification) instruments. Y

** The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

PROOF AND REVIEW

TABLE 3.3-11

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - CONTROL			
1	74'	Essential Chiller Rm. - Train A			24/0
2	74'	Essential Chiller Rm. - Train B			21/0
3A	74'	Cable Shaft - Train A			1/0
~~43~~B	74'	Cable Shaft - Train B			1/0
86A	74-156'4"	Deadspace Compartment - Train A	0/1		0/3
86B	74-156'4"	Deadspace Compartment - Train B	0/1		0/3
4A	100'	Cable Shaft - Train A			1/0
4B	100'	Cable Shaft - Train B			1/0
5A	100'	ESF Switchgear Room Train A			0/10
5B	100'	ESF Switchgear Room Train B			0/10
6A	100'	DC Equipment Rm. - Tr. A (Channel C)			2/0
6B	100'	DC Equipment Rm. - Tr. B (Channel D)			2/0
7A	100'	DC Equipment Rm. - Tr. A (Channel A)			2/0
7B	100'	DC Equipment Rm. - Tr. B (Channel B)			2/0
8A	100'	Battery Rm. - Tr. A (Channel C)	0/2		0/2
8B	100'	Battery Rm. - Tr. B (Channel D)	0/2		0/2

PROOF AND REVIEW

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
9A	100'	Battery Rm - Train A (Channel A)	0/2		0/2
9B	100'	Battery Rm - Train B (Channel B)	0/2		0/2
10A	100'	Remote Shutdown Rm. Train A			1/0
10B	100'	Remote Shutdown Rm. Train B			1/0
11A	120'	Cable Shaft - Train A			1/0
11B	120'	Cable Shaft - Train B			1/0
14	120'	Lower Cable Spreading Rm.			
		a. System 1	0/1		0/6
		b. System 2	0/1		0/6
		c. System 3	0/1		0/8
		d. System 4	0/1		0/8
		e. System 5	0/1		0/8
		f. System 6	0/1		0/8
15A	140'	Cable Shaft Tr. A			1/0
15B	140'	Cable Shaft Tr. B			1/0
17	140'	Control Rm - MCB's & Relay Cabinets			97/0
18A	160'	Cable Shaft - Train A			1/0
18B	160'	Cable Shaft - Train B			1/0

PROOF AND REVIEW

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
20	160'	Upper Cable Spreading Rm.			
		System 1	0/1		0/12
		System 2	0/1		0/8
		System 3	0/1		0/8
		System 4	0/1		0/8
		System 5	0/1		0/8
		BUILDING - DIESEL GENERATOR			
21A	100'	Diesel Generator - Tr. A	0/3	0/4	
21B	100'	Diesel Generator - Tr. B	0/3	0/4	
22A	100'	Diesel Generator Control Rm. Tr. A			1/0
22B	100'	Diesel Generator Control Rm. Tr. B			1/0
24A	115'	Combustion Air Intake Rm. Tr. A			1/0
24B	115'	Combustion Air Intake Rm. Tr. B			1/0
23A	131'	Fuel Oil Day Tank Tr. A	0/1		
23B	131'	Fuel Oil Day Tank Tr. B	0/1		
25A	131'	Exhaust Silencer Rm. Tr. A		3/0	
25B	131'	Exhaust Silencer Rm. Tr. B		3/0	
		BUILDING - FUEL			
28	100'	Spent Fuel Pool Cooling and Cleanup Pump Areas			3/0

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FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		Building Auxiliary			
88A	51'-6"	West Corridors			6/0
88B	51'-6"	East corridors			6/0
32A	51' - 6"	LPSI Pump Rm. - Tr. A			0/2
32B	51' - 6"	LPSI Pump Rm. - Tr. B			0/2
34A	70'	ECW Pump Rm. - Tr. A			2/0
34B	70'	ECW Pump Rm. - Tr. B			2/0
35A	70'	Shutdown Cooling Ht. x Chgr. Tr. A			4/0
35B	70'	Shutdown Cooling Ht. x Chgr. Tr. B			4/0
37C	70'+88'	Piping Penetration Rm. - Tr. A			~~[illegible]~~ 5/0
37D	70'+88'	Piping Penetration Rm. - Tr. B			~~3/0~~ 4/0
37B	70'	Corridors - East			~~5/0~~ 11/0
37A	70'	Corridors - West			~~8/0~~ 11/0
39A	88'	Pipeways - Tr. A			~~[illegible]~~ 8/0
39B	88'	Pipeways - Tr. B			~~12/0~~ 8/0
42A	100'	Elect. Penetration Rm. Tr. A (Channel C)	0/1		0/25

PROOF AND REVIEW

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
42B	100'	Elect. Penetration Rm. Tr. B (Channel B)	0/1		0/24
42C	100'	Corridors - East + Southeast	0/2		3/35
42D	100'	Corridor - west	0/1		2/29
46A	100'	Charging Pump + valve Gallery Rm - TR A			0/3
46B	100'	" " - TR B			0/3
46E	100'	" " - TR E			0/3
47A	120'	Elect. Penetration Rm. Tr. A (Channel A)	0/1		0/28
47B	120'	Elect. Penetration Rm. Tr. B (Channel D)	0/1		0/24
48	120'	ECW Surge Tanks Corridor - TR A+B			3/0
50B	120'	valve Gallery			1/0
51B	120'	Spray Chemical Storage Tk Rm			1/0
52A	120'	Central Corridor - west	0/1		5/17
52D	120'	Central Corridor - East	0/1		7/17
54	120'	Reactor Trip Switchgear Rm	1/0		6/0
56B	140'	Storage and Elect. Equip. Rm - East			6/0
57I	140'	Clothing Issue and men's Locker Rm			5/0
57J	140'	women's Locker, clean storage and Lunch Rms			7/0
57N	140'	Corridor area			4/0

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		Building - Containment **			
66A+66B	100'	Southwest and Southeast Perimeter	1/0		
67A+67B	100'	northwest and northeast Perimeter	1/0		
66A	120'	Southwest Perimeter	1/0		
66B	120'	Southeast Perimeter	1/0		
67A+67B	120'	northwest and northeast Perimeter	1/0		
63A	120'	no. 1 RCPS and SG area			6/0
63B	120'	no. 2 RCPS and SG area			6/0
6A[illegible]B, 67A+67B	140'	Southwest, Southeast, northwest and northeast Perimeters	1/0		
63A	140'	no. 1 RCPS and SG area			5/0
63B	140'	no. 2 RCPs and SG area			5/0
70	140'	Refueling Pool and Canal area			4/0
71A	140'	north preaccess normal AFU area			2/0
71B	140'	South preaccess normal AFU area			2/0

PROOF AND REVIEW

TABLE 3.3-11 (Continued)

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FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		MAIN STEAM SUPPORT STRUCTURE			
72	80'	Turbine Driven Aux. Feedpump Rm.		0/3	
73	80'	Motor Driven Aux. Feedpump Rd.		1/1	
74A	100', 120' + 140'	Main Steam Isol + Dump valve area - north		~~6/0~~ 0/6	
74B	100', 120' + 140'	Main Steam Isol + Dump valve area - South		~~6/0~~ 0/6	
		Outside Areas			
83		Condensate Storage tank Pump House		2/0	
84A		Spray Pond Pump House - TR. A		1/0	
84B		Spray Pond Pump House - TR. B		1/0	

PROOF AND REVIEW

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENT

FPER ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS		
			HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)

(x/y): x is the number of early warning fire detection and notification only instruments.

y is the number of actuation of fire suppression systems and early warning fire detection and notification instruments.

* The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

X IS THE NUMBER OF INSTRUMENTS ASSOCIATED WITH EARLY DETECTION AND NOTIFICATION ONLY

Y IS THE NUMBER OF INSTRUMENTS ASSOCIATED WITH ACTUATION OF FIRE SUPPRESSION SYSTEMS AND EARLY FIRE DETECTION AND NOTIFICATION

PROOF AND REVIEW

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The loose-part detection system shall be OPERABLE with all sensors specified in Table 3.3-12.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. a CHANNEL CHECK at least once per 24 hours,

b. a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. a CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.3-12

LOOSE PARTS SENSOR LOCATIONS

INSTRUMENT NO.	LOCATION
JSVNYE - 1	UPPER VESSEL A (STUD BOLTS)
JSVNYE - 2	UPPER VESSEL B (STUD BOLTS)
JSVNYE - 3	LOWER VESSEL A (INCORE NOZZLE)
JSVYNE - 4	LOWER VESSEL B (INCORE NOZZLE)
JSVYNE - 5	SG-1A (HOT LEG)
JSVYNE - 6	SG-1B (COLD LEG 1A)
JSVYNE - 7	SG-2A (HOT LEG)
JSVYNE - 8	SG-2B (COLD LEG 2A)

PROOF AND REVIEW

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel; or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8, why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS RADWASTE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12	1	~~*~~ #	35
b. Flow Rate Monitor	1	~~*~~ #	36
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	39
b. Oxygen Monitor	1	**	39

TABLE 3.3-1[illegible] (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3.	CONDENSER EVACUATION SYSTEM			
	a. Noble Gas Activity Monitor #RU-141	1	1, 2, 3, 4	37
	b. Iodine Sampler	1	1, 2, 3, 4	40
	c. Particulate Sampler	1	1, 2, 3, 4	40
	d. Flow Rate Monitor	1	1, 2, 3, 4	36
	e. Sampler Flow Rate Measuring Device	1	1, 2, 3, 4	36
4.	PLANT VENT SYSTEM			
	a. Noble Gas Activity Monitor #RU-143 and #RU-144	1	*	37
	b. Iodine Sampler	1	*	40
	c. Particulate Sampler	1	*	40
	d. Flow Rate Monitor	1	*	36
	e. Sampler Flow Rate Measuring Device	1	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. FUEL BUILDING VENTILATION SYSTEM			
a. Noble Gas Activity Monitor #RU-145	1	##	37
b. Iodine Sampler	1	##	40
c. Particulate Sampler	1	##	40
d. Flow Rate Monitor	1	##	36
e. Sampler Flow Rate Measuring Device	1	##	36

PROOF AND REVIEW

DURING WASTE GAS RELEASE

WITH IRRIDIATED FUEL IN THE STORAGE POOL, OR IN MODES 1, 2, 3 OR 4

AUG 1984
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NUCLEAR OPERATIONS LICENSING

TABLE 3.3-13 (Continued)

TABLE NOTATION

STET

~~* At all times.~~

~~** During GASEOUS RADWASTE SYSTEM operation.~~

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this GASEOUS RADWASTE SYSTEM may continue for up to 14 days provided grab samples are collected at least once per 8 hours and are analyzed within the following 4 hours for the "on service" gas decay tank. ~~With both channels inoperable, be in at least HOT STANDBY within 6 hours.~~

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 37 WITH THE NUMBER OF CHANNELS OPERABLE LESS THAN THE REQUIRED BY THE MINIMUM CHANNELS OPERABLE REQUIREMENT, RELEASES VIA THIS PATHWAY MAY CONTINUE FOR UP TO 30 DAYS, PROVIDED THE ACTIONS OF (A) OR (B) ARE PERFORMED
COMPLETE THE ACTIONS OF A OR B
A. INITIATE THE PREPLANNED ALTERNATE METHOD OF SAMPLING THE APPROPRIATE PRAMETER(S)
B. TAKE GRAB SAMPLES ATLEAST ONCE PER 12 HOURS AND ANALYZE FOR GROSS ACTIVITY WITHIN 24 HRS

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS RADWASTE SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1), P###	~~*~~ #
b. Flow Rate Monitor	P	N.A.	R	Q, P###	~~*~~ #
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (sequential)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (sequential)	D	N.A.	Q(5)	M	**

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	1, 2, 3, 4
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Flow Rate Monitor	D	N.A.	R	Q	1, 2, 3, 4
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	1, 2, 3, 4
4. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
5. FUEL BUILDING VENTILATION SYSTEM					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	~~*~~ ##
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	~~*~~ ##
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	~~*~~ ##
d. Flow Rate Monitor	D	N.A.	R	Q	~~*~~ ##
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	~~*~~ ##

PROOF AND REVIEW

RECEIVED AUG 1984 NUCLEAR OPERATIONS LICENSING

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

DURING WASTE GAS RELEASE
WITH IRRADIATED FUEL IN THE STORAGE POOL OR MODES 1,2,3 OR 4
FUNCTIONAL TEST SHOULD CONSIST OF BUT NOT BE LIMITED TO A VERIFICATION OF SYSTEM ISOLATION CAPABILITY BY THE INSERTION OF A SIMULATED ALARM CONDITION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

1. Four high pressure turbine stop valves.
2. Four high pressure turbine control valves.
3. Six low pressure turbine reheat stop valves.
4. Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

---

*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

PROOF AND REVIEW

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be $\geq$ 25% indicated wide range level at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 25% indicated wide range level at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 4000 gpm at least once per 12 hours.

PROOF AND REVIEW

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum steady-state water level of greater than or equal to 27% indicated level (425 cubic feet) and a maximum steady-state water level of less than or equal to 56% indicated level (948 cubic feet) and at least two groups of pressurizer heaters capable of being powered from Class 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

REACTOR COOLANT SYSTEM

PRESSURIZER

AUXILIARY SPRAY

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With none of the above required auxiliary spray valves OPERABLE, restore at least one valve to operable within the next 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 The auxiliary spray valves shall be cycled during normal plant cooldown at least once per 18 months.

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3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{cold} above 210°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.8.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C–1	None	N. A.	N. A.	N. A.	N. A.
	C–2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N. A.	N. A.
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N. A.	N. A.
	C–3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C–1	None	N. A.	N. A.
			Some S. G.s C–2 but no additional S. G. are C–3	Perform action for C–2 result of second sample	N. A.	N. A.
			Additional S. G. is C–3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere (gaseous and particulate) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table (4.3-3),

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of ~~2235~~ 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve or within 72 hours following a system response to an Engineered Safety Feature actuation signal.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

*TESTING PER SPECIFICATION 4.4.5.2.2.d IS NOT APPLICABLE DUE TO POSITIVE INDICATION OF VALVE POSITION IN THE CONTROL ROOM

#1. LEAKAGE RATES LESS THAN OR EQUAL TO 1.0 GPM ARE CONSIDERED ACCEPTABLE

#2. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED ACCEPTABLE IF THE LATEST MEASURED RATE HAS NOT EXCEEDED THE RATE DETERMINED BY PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

#3. LEAKAGE RATES GREATER THAN 1.0 GPM BUT LESS THAN OR EQUAL TO 5.0 GPM ARE CONSIDERED UNACCEPTABLE IF THE LATEST MEASURED RATE EXCEEDED THE RATE DETERMINED BY THE PREVIOUS TEST BY AN AMOUNT THAT REDUCES THE MARGIN BETWEEN MEASURED LEAKAGE RATE AND THE MAXIMUM PERMISSIBLE RATE OF 5.0 GPM BY 50% OR GREATER.

#4. LEAKAGE RATES GREATER THAN 5.0 GPM ARE CONSIDERED UNACCEPTABLE.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE	DESCRIPTION
1) SIV 237	LOOP 1A RC/SI CHECK
2) SIV 247	LOOP 1B RC/SI CHECK
3) SIV 217	LOOP 2A RC/SI CHECK
4) SIV 227	LOOP 2B RC/SI CHECK
5) SIV 235	LOOP 1A SIT CHECK
6) SIV 245	LOOP 1B SIT CHECK
7) SIV 215	LOOP 2A SIT CHECK
8) SIV 225	LOOP 2B SIT CHECK
9) SIV 542	LOOP 1A SI HEADER CHECK
10) SIV 543	LOOP 1B SI HEADER CHECK
11) SIV 540	LOOP 2A SI HEADER CHECK
12) SIV 541	LOOP 2B SI HEADER CHECK
13) SIV 522	LOOP 1 HP LONG TERM RECIRCULATION CHECK
14) SIV 523	LOOP 1 HP LONG TERM RECIRCULATION CHECK
15) SIV 532	LOOP 2 HP LONG TERM RECIRCULATION CHECK
16) SIV 533	LOOP 2 HP LONG TERM RECIRCULATION CHECK
17) UV 651 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
18) UV 652 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION
19) UV 653 *,#	LOOP 1 SHUTDOWN COOLING ISOLATION
20) UV 654 *,#	LOOP 2 SHUTDOWN COOLING ISOLATION

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3/4.4.6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm

*Limit not applicable with T_{cold} less than or equal to 250°F.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with T_{cold} less than or equal to 250°F

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3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

---

*With T_{cold} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

MODES 1, 2, 3, 4, and 5:

d. With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4.(a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,
2. Fuel burnup by core region,
3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	(a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	(b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. ONE SAMPLE IS SUFFICIENT IF PLANT HAS GONE THROUGH A SCRAM CONDITION OR IF TRANSIENT IS COMPLETE IN 6 HOURS	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

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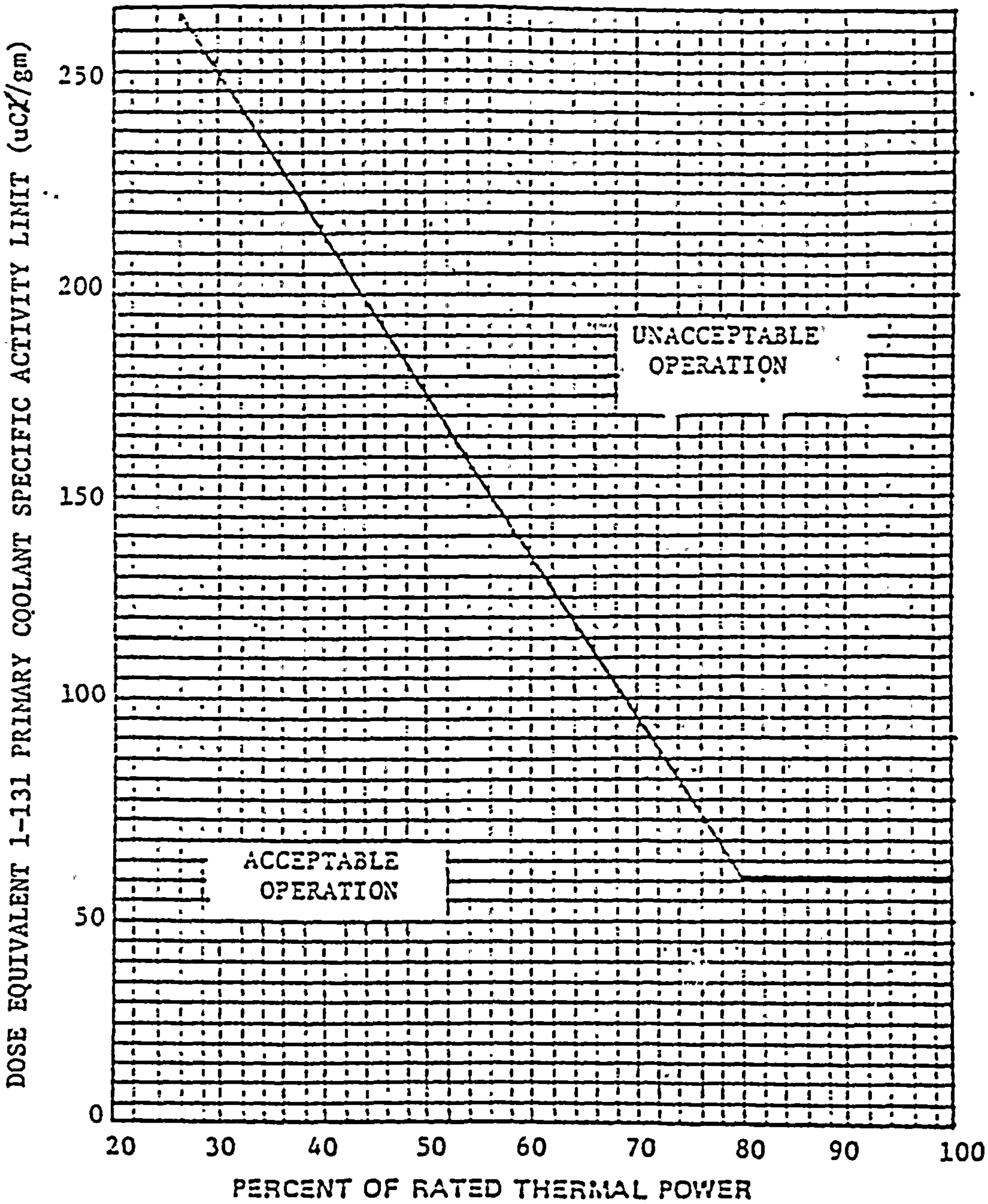

FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 20°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

DATE	SUBJECT	
DATE		JOB NO

THIS GRAPH WILL BE COMING IN

ABOUT A WEEK.

4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	38°	1.5	8 - 10
2	43°	1.5	Standby
3	137°	1.5	4 - 5
4	142°	1.5	Standby
5	230°	1.5	12 - 15
6	310°	1.5	18 - 24

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PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 417 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 255°F during cooldown

b. 295°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 255°F.

b. Heatup with the RCS temperature less than or equal to 295°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

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3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 0, October 27, 1971.

REACTOR COOLANT SYSTEM

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vents paths inoperable, STARTUP and/or POWER OPERATIONS may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent valve through at least one complete cycle of full travel from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between ~~28%~~ (1802 cubic feet) and ~~72%~~ (1914 cubic feet) level as read on narrow range indication) (E.G. BETWEEN 28% AND 72%),

c. A boron concentration between ~~4000~~ 2000 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**

f. Nitrogen vent valves are capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

---

†With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 60% wide range indication (1415 cubic feet) and 72% narrow range indication (1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 39% wide range indication (962 cubic feet) and 72% narrow range indication (1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

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2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.
2000

c. At least once per 31 days when the RCS pressure is above 700 psig, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 700 psig with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

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3/4.5.2 ECCS SUBSYSTEMS - T_{cold} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1750 psia.

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4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIHV-604	1. HOT LEG INJECTION	1. SHUT
2. SIHV-321	2. HOT LEG INJECTION	2. SHUT
3. SIHV-609	3. HOT LEG INJECTION	3. SHUT
4. SIHV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

2. Verifying that a minimum total of 464 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

3. Verifying that when a representative sample of 0.055 ± 0.001 lb. of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on (SIAS and RAS) test signal(s).

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a. High pressure safety injection pump.

   b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI and LPSI pump minimum bypass recirculation flow line isolation valves close, and the LPSI pumps stop.

f. By verifying that each of the following pumps develops the differential indicated pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1830 psid.

2. Low pressure safety injection pump greater than or equal to 195 psid.
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g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number	Hot Leg Injection Valve Number
1. SI 617, SI 616	1. SI 615, SI 306	1. SI-604
2. SI 627, SI 626	2. SI 625, SI 307	2. SI-609
3. SI 637, SI 636	3. SI 635	3. SI-321
4. SI 647, SI 646	4. SI 645	4. SI-331

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System - Single Pump

1. Injection Leg 1A, equal to 277 ± 5 gpm
2. Injection Leg 1B, equal to 277 ± 5 gpm
3. Injection Leg 2A, equal to 277 ± 5 gpm
4. Injection Leg 2B, equal to 277 ± 5 gpm

THE SUM OF THE INJECTION LINE FLOW RATES, EXCLUDING THE HIGHEST FLOWRATE, IS GREATER THAN OR EQUAL TO 816 GPM

LPSI System Single Pump

1. Injection ~~Leg~~ LOOP 1, total flow equal to 4900 ± 100 gpm
2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.
3. Injection ~~Leg~~ LOOP 2, total flow equal to 4900 ± 100 gpm
4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.

Simultaneous Hot Leg and Cold Leg Injection - Single Pump

1. Hot Leg, flow equal to 545 ± 20 gpm
2. Cold Leg, flow equal to 545 ± 20 gpm

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3/4.5.3 ECCS SUBSYSTEMS - T_{cold} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1750 psia.

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3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank (RWT) shall be OPERABLE with:

a. A minimum borated water volume as specified in Figure 3.1-2 of Specification 3.1.2.5, and

b. A boron concentration between 4000 and 4400 ppm of boron, and

c. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is outside the 60°F to 120°F range.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* ~~not~~ IN TABLE 3.6-0 (NOT capable of being closed by OPERABLE** containment automatic isolation valves and required to be closed during accident conditions) are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, ~~except as provided in Table 3.6-1 of Specification 3.6.3.~~

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type .. or B test, by leak rate testing the seal with gas at P_a 49.2 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

** A CLOSED, ISOLATED, OR BLANK FLANGED VALVE IS CONSIDERED OPERABLE FOR CONTAINMENT ISOLATION

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TABLE 3.6-~~1~~ 0 ~~(Continued~~

CONTAINMENT ISOLATION ~~VALVES~~. PENETRATIONS

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VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A ~~E~~. NORMALLY CLOSED/POST ACCIDENT CLOSED VALVES	
SG-V-603	1	N_2 blanket supply/N_2 vent	N.A.
SG-V-611	3	N_2 blanket supply/N_2 vent	N.A.
~~SG-HV-184~~	~~1~~	~~Main steam atmospheric dump~~	~~N.A.~~
~~SG-HV-178~~	~~2~~	~~Main steam atmospheric dump~~	~~N.A.~~
~~SG-HV-185~~	~~3~~	~~Main steam atmospheric dump~~	~~N.A.~~
~~SG-HV-179~~	~~4~~	~~Main steam atmospheric dump~~	~~N.A.~~
DW-V 061*	6	Containment demineralized water stations	N.A.
DW-V 062*	6	Containment demineralized water stations	N.A.
FP-V 089	7	Fire protection containment	N.A.
SI-V 463*	28	Safety injection drain from drain tank	N.A.
CH-V 854*	41	Chemical addition unit to regenerative heat exchanger	N.A.
PC-V 070	50	Fuel pool cooling	N.A.
PC-V 071	50	Fuel pool cooling	N.A.
PC-V 075	51	Refueling pool cleanup	N.A.
PC-V 076	51	Refueling pool cleanup	N.A.
IA-V 072*	59	Containment service air utility station	N.A.

*May be opened on an intermittent basis under administrative control.

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TABLE 3.6-~~1~~ 0 (Continued

CONTAINMENT ISOLATION ~~VALVES~~ PENETRATIONS

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		B ~~F~~. SAFETY/RELIEF VALVES (Continued)	
SI-PSV 151	23	Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B	N.A.
SI-PSV 140	24	Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B	N.A.
SI-PSV 189	26	From shutdown cooling RC loop 2	N.A.
SI-PSV 179	27	From shutdown cooling RC loop 1	N.A.
SI-PSV 474	28	Safety injection drain relief	N.A.

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TABLE 3.6-~~1~~ 20 (Continued

CONTAINMENT ISOLATION ~~VALVES~~ PENETRATION

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		C. ~~B.~~ CHECK VALVES	
FP-V 090	7	Containment fire protection	N.A.
~~SG-V 003~~	~~8~~	~~Steam generator feedwater~~	~~N.A.~~
~~SG-V 007~~	~~8~~	~~Steam generator feedwater~~	~~N.A.~~
~~SG-V 005~~	~~10~~	~~Steam generator feedwater~~	~~N.A.~~
~~SG-V 006~~	~~10~~	~~Steam generator feedwater~~	~~N.A.~~
~~SG-V 642~~	~~11~~	~~Feedwater downcomer~~	~~N.A.~~
~~SG-V 652~~	~~11~~	~~Feedwater downcomer~~	~~N.A.~~
~~SG-V 653~~	~~12~~	~~Feedwater downcomer~~	~~N.A.~~
~~SG-V 693~~	~~12~~	~~Feedwater downcomer~~	~~N.A.~~
~~SI-V 113~~	~~13~~	~~HPSI to RC loop 2A~~	~~N.A.~~
~~SI-V 123~~	~~14~~	~~HPSI to RC loop 2B~~	~~N.A.~~
~~SI-V 133~~	~~15~~	~~HPSI to RC loop 1A~~	~~N.A.~~
~~SI-V 143~~	~~16~~	~~HPSI to RC loop 1B~~	~~N.A.~~
~~SI-V 114~~	~~17~~	~~LPSI to RC loop 2A~~	~~N.A.~~
~~SI-V 124~~	~~18~~	~~LPSI to RC loop 2B~~	~~N.A.~~
~~SI-V 134~~	~~19~~	~~LPSI to RC loop 1A~~	~~N.A.~~
~~SI-V 144~~	~~20~~	~~LPSI to RC loop 1B~~	~~N.A.~~
~~SI-V 164~~	~~21~~	~~Shutdown cooling heat exchanger 1 to containment spray header 1~~	N.A.
~~SI-V 165~~	~~22~~	~~Shutdown cooling heat exchanger 2 to containment spray header 2~~	N.A.
GA-V 015	29	N_2 to steam generator and reactor drain tank	N.A.

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TABLE 3.6-~~2~~9 (Continued

CONTAINMENT ISOLATION ~~VALVES~~ PENETRATIONS

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		C. ~~8~~. CHECK VALVES (Continued)	
GA-V 011	30	N_2 to SI tanks	N.A.
IA-V 021	31	Service air to reactor containment instrument air header	N.A.
NC-V 118	33	NC water to RCP motor bearing lube oil and air coolers	N.A.
HP-V 002	38	H_2 recombiner return to containment	N.A.
HP-V 004	39	H_2 recombiner return to containment	N.A.
~~CH-VM 70~~	~~41~~	~~Regenerative heat exchanger to~~ RC loop 2A	~~N.A.~~
CH-V 494	45	Makeup to reactor drain tank	N.A.
IA-V 073	59	Containment service air utility station	N.A.
WC-V 039	60	Normal chilled water to containment ACU	N.A.
~~SI-V 533~~	~~67~~	~~Long term recirculation loop 2~~	~~N.A.~~
~~CH-V 835~~	~~72~~	~~RC pump seal injection water to RCP 1A, 1B, 2A, 2B~~	~~N.A.~~
~~AF-V 079~~	~~75~~	~~Steam generator 1 auxiliary feedwater~~	~~N.A.~~
~~AF-V 080~~	~~76~~	~~Steam generator 2 auxiliary feedwater~~	~~N.A.~~
~~SI-V 523~~	~~77~~	~~Long term recirculation loop 1~~	~~N.A.~~

N.A. - Actuation time not applicable.

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TABLE 3.6-D

CONTAINMENT ISOLATION PENETRATIONS

VALVE NUMBER	PENETRATION NUMBER	FUNCTION
		D. FLANGES, HATCHES AND SPARE PENETRATIONS
NA	5	SPARE
NA	53	REFUELING POOL FLANGE
NA	58	PERSONNEL AIR LOCK ILRT FLANGE
NA	62B	CONTAINMENT PRESSURE ILRT FLANGE
NA	62C	CONTAINMENT PRESSURE ILRT FLANGE
NA	64	SPARE
NA	65	SPARE
NA	66	SPARE
NA	68	SPARE
NA	69	SPARE
NA	70	SPARE
NA	71	SPARE
NA	80	SPARE
NA	81	SPARE
NA	L-1	PERSONNEL AIRLOCK
NA	L-2	EQUIPMENT HATCH
NA	L-3	EMERGENCY AIRLOCK

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TABLE 3.6-C

CONTAINMENT ISOLATION PENETRATIONS

VALVE NUMBER	PENETRATION NUMBER	FUNCTION
		C. CHECK VALVES
FP-V 090	7	Containment fire protection
GA-V 015	29	N_2 to steam generator and reactor drain tank
GA-V 011	30	N_2 to SI tanks
IA-V 021	31	Service air to reactor containment instrument air header
NC-V 118	33	NC water to RCP motor bearing lube oil and air coolers
HP-V 002	38	H_2 recombiner return to containment
HP-V 004	39	H_2 recombiner return to containment
CH-V 494	45	Makeup to reactor drain tank
IA-V 073	59	Containment service air utility station
WC-V 039	60	Normal chilled water to containment ACU

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.10% by weight of the containment air per 24 hours at P_a, 49.2 psig, or

2. Less than or equal to L_t, 0.05% by weight of the containment air per 24 hours at a reduced pressure of P_t, 24.6 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 210°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a 49.2 psig or at P_t 24.6 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every REFUELING OR 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at, WHICHEVER COMES FIRST, which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

   2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 49.2 psig, or P_t, 24.6 psig.

d. Type B and C tests shall be conducted with gas at P_a, 49.2 psig, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Specifications 4.6.1.7.3 and 4.6.1.7.4.

f. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3.

g. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 49.2 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or

2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage to be less than or equal to 0.01 L_a when determined with the volume between the door seals pressurized to greater than or equal to 14.5 ± 0.5 psig, for at least 15 minutes,

SURVEILLANCE REQUIREMENTS (Ccontinued)

b. By conducting overall air lock leakage tests at not less than P_a, 49.2 psig, and verifying the overall air lock leakage rate is within its limit:

   1. At least once per 6 months#, and

   2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR Part 50.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any five of the following locations and shall be determined at least once per 24 hours:

Location

a. Elevation 85'0"

b. Elevation 85'0"

c. Elevation 126'0"

d. Elevation 126'0"

e. Elevation 145'0"

f. Elevation 188'0"

g. Elevation 188'0"

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, ~~restore the structural integrity to within the limits within 24 hours or~~ PERFORM AN ENGINEERING EVALUATION OF THE CONTAINMENT TO DEMONSTRATE ITS STRUCTURAL INTEGRITY WITHIN 72 HOURS; OTHERWISE be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Tendons The containment tendons' structural integrity shall be demonstrated at the end of 1, 3 and 5 years following the initial containment structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections the maximum first year lift off forces shall be decreased by the amount X log t, and the minimum lift off forces shall be decreased by the amount Y log t where t is the time interval in years from initial tensioning of the tendon to the current testing date. If no abnormal degradation of the containment tendons is detected during the 1-, 3-, and 5-year inspections, the number of tendons checked for lift-off force may be reduced to 2% of the population of each group (hoop and inverted U) with a minimum of three tendons fc each group. The sample size from any group need not exceed five. For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (hoop and inverted U) may be kept unchanged after the initial selection.

b. Removing one wire from one U tendon and one hoop tendon checked for lift-off force and determining that over the entire length of the removed wire or strand that:

   1. The tendon wires or strands are free of corrosion, cracks, and damage.

   2. There are no changes in the presence or physical appearance of the sheathing filler grease.

   3. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

4.6.1.6.2 <u>End Anchorages and Adjacent Concrete Surfaces</u> The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.6.3 <u>Containment Surfaces</u> The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.4 <u>Reports</u> Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

PROOF AND REVIEW

TABLE 4.6-1

TENDON SURVEILLANCE - FIRST YEAR

Tendon No.	Visual Inspection	Monitor Forces	Detension Tendon	Remove Wire	Test Wire
V32	X	X	No	No	No
V43	X	X	No	No	No
V62	X	X	X	X	X
V75*	X	X	A	A	A
H13-007*	X	X	X	X	X
H13-021	X	X	No	No	No
H21-037	X	X	No	No	No
H21-044	X	X	No	No	No
H32-016	X	X	No	No	No
H32-030	X	X	A	A	A

Notes:

1. "X" means the tendon shown shall be inspected for the stated requirements during this surveillance.
2. "A" means the tendon shown shall be inspected for the stated requirements during the next or second surveillance.
3. "No" means that inspection is not required for that tendon.
4. "*" means control tendon.

PROOF AND REVIEW

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TABLE 4.6-2

TENDON LIFT-OFF FORCE
U-TENDONS

PROOF AND REVIEW

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	Y
V3	Shop	1425	1304	22.9	32.4
	Field	1402	1282	22.7	32.2
V8	Shop	1513	1388	23.1	32.8
	Field	1466	1344	22.9	32.5
V11	Shop	1460	1338	22.9	32.5
	Field	1438	1317	22.9	32.5
V14	Shop	1525	1399	23.3	32.9
	Field	1491	1366	23.1	32.7
V15	Shop	1459	1337	22.9	32.5
	Field	1436	1315	22.8	32.3
V18	Shop	1476	1352	23.0	32.6
	Field	1453	1330	22.9	32.5
V21	Shop	1509	1385	23.0	32.6
	Field	1462	1341	22.8	32.3
V26	Shop	1402	1283	22.7	32.1
	Field	1471	1348	23.0	32.6
V29	Shop	1533	1408	23.1	32.8
	Field	1533	1408	23.1	32.8
V32	Shop	1463	1343	22.8	32.3
	Field	1510	1386	23.0	32.6
V36	Shop	1475	1354	22.9	32.4
	Field	1533	1408	23.1	32.8
V37	Shop	1510	1386	23.0	32.6
	Field	1486	1364	22.9	32.5
V40	Shop	1543	1416	23.3	32.9
	Field	1472	1350	22.9	32.5
V45	Shop	1457	1336	22.9	32.4
	Field	1526	1400	23.2	32.9
V47	Shop	1437	1315	22.9	32.5
	Field	1506	1379	23.3	32.9
V53	Shop	1466	1344	22.9	32.5
	Field	1513	1388	23.1	32.8
V54	Shop	1514	1389	23.1	32.8
	Field	1445	1324	22.8	32.3
V55	Shop	1457	1335	22.9	32.5
	Field	1469	1346	23.0	32.5
V62	Shop	1475	1354	22.9	32.4
	Field	1486	1364	22.9	32.5
V67	Shop	1442	1320	22.9	32.4
	Field	1476	1352	23.0	32.6
V72	Shop	1510	1386	23.0	32.6
	Field	1510	1386	23.0	32.6
V75	Shop	1527	1402	23.1	32.8
	Field	1504	1380	23.0	32.6
V83	Shop	1458	1334	23.0	32.6
	Field	1389	1270	22.7	32.1
V86	Shop	1505	1378	23.3	32.9
	Field	1436	1314	22.9	32.5

TABLE 4.6-2 (Continued)

TENDON LIFT-OFF FORCE
HOOP TENDONS

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	Y
H21-001	Shop	1516	1384	29.8	42.3
	Field	1505	1373	29.8	42.3
H21-003	Shop	1487	1356	29.7	42.2
	Field	1499	1367	29.8	42.3
H21-006	Shop	1422	1292	29.6	42.0
	Field	1513	1377	30.1	42.7
H13-008	Shop	1441	1309	29.7	42.2
	Field	1475	1341	29.9	42.4
H32-009	Shop	1517	1382	30.0	42.5
	Field	1493	1360	29.8	42.3
H32-010	Shop	1480	1345	30.0	42.5
	Field	1480	1345	30.0	42.5
H21-011	Shop	1505	1371	29.9	42.4
	Field	1458	1328	29.7	42.1
H32-012	Shop	1434	1303	29.7	42.2
	Field	1480	1345	30.0	42.5
H21-013	Shop	1470	1339	29.7	42.2
	Field	1470	1339	29.7	42.2
H13-014	Shop	1473	1339	30.0	42.5
	Field	1450	1317	29.8	42.3
H32-015	Shop	1517	1382	30.0	42.5
	Field	1505	1371	29.9	42.4
H32-016	Shop	1411	1282	29.6	42.0
	Field	1457	1324	29.8	42.3
H13-019	Shop	1445	1315	29.6	42.0
	Field	1445	1315	29.6	42.0
H13-021	Shop	1515	1380	30.0	42.5
	Field	1491	1358	29.8	42.3
H21-021	Shop	1445	1315	29.6	42.0
	Field	1445	1315	29.6	42.0
H32-023	Shop	1470	1339	29.7	42.2
	Field	1493	1360	29.8	42.3
H13-025	Shop	1412	1284	29.4	41.8
	Field	1505	1371	29.9	42.4
H32-026	Shop	1389	1261	29.5	41.9
	Field	1480	1345	30.0	42.5
H21-028	Shop	1473	1339	30.0	42.5
	Field	1450	1317	29.8	42.3
H13-030	Shop	1428	1297	29.7	42.2
	Field	1428	1297	29.7	42.2
H32-033	Shop	1470	1339	29.7	42.2
	Field	1505	1371	29.9	42.4
H21-037	Shop	1505	1371	29.9	42.4
	Field	1446	1317	29.6	42.0
H32-040	Shop	1417	1286	29.7	42.1
	Field	1417	1286	29.7	42.1

TABLE 4.6-2 (Continued)

TENDON LIFT-OFF FORCE
HOOP TENDONS

PROOF AND REVIEW

TENDON NUMBER	END	FIRST YEAR MAXIMUM (kips)	FIRST YEAR MINIMUM (kips)	X	Y
H32-041	Shop	1475	1352	23.0	32.5
	Field	1522	1395	23.2	32.9
H21-042	Shop	1507	1381	23.1	32.8
	Field	1530	1403	23.3	32.9
H13-043	Shop	1421	1302	22.7	32.1
	Field	1492	1368	23.0	32.6
H32-044	Shop	1472	1349	23.0	32.5
	Field	1530	1403	23.3	32.9
H21-045	Shop	1510	1385	23.1	32.7
	Field	1545	1418	23.3	32.9
H21-046	Shop	1537	1410	23.3	32.9
	Field	1548	1421	23.3	33.0
H13-047	Shop	1439	1319	22.7	32.2
	Field	1486	1363	23.0	32.5

DELETE

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 42-inch containment purge supply and exhaust isolation valve shall be sealed closed.
b. The 8-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 3000 hours per 365 days.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for more than 3000 hours per 365 days, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specifications 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

OR ISOLATE THE PENETRATIONS

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 42-inch containment purge supply and exhaust isolation valves shall be verified to be sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 8-inch purge supply or exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 42-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.4 At least once per 92 days each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.01 L_a when pressurized to P_a.

PROOF AND REVIEW

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a containment spray actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is positioned to take suction from the RWT on a containment spray actuation (CSAS) test signal.

b. By verifying that each pump develops an indicated differential pressure of greater than or equal to 270 psid at greater than or equal the minimum allowable recirculation flowrate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water to the 60 inch level in the containment spray header (≥115 foot level).

d. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) and recirculation actuation (RAS) test signal.

2. Verifying that upon a recirculation actuation test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

*Only when shutdown cooling is not in operation.

3. Verifying that each spray pump starts automatically on a containment spray actuation (CSAS) test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

..12.

PROOF AND REVIEW

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. An spray chemical addition tank containing a level of between 90% and 100% (816 and 896 gallons) of between 33% and 35% by weight N_2H_4 solution, and

b. Two spray chemical addition pumps each capable of adding N_2H_4 solution from the spray chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the N_2H_4 solution by chemical analysis.

c. By verifying that on recirculation flow, each spray chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5.

d. At least once per 18 months, during shutdown, by

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) test signal, and

2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 5 years by verifying each solution flow rate from the following drain connections in the iodine removal system:

1. SIA-V253 Pump discharge line 0.63 ± 0.02 gpm.

2. SIB-V254 Pump discharge line 0.63 ± 0.02 gpm.

PROOF AND REVIEW

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE* with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable; maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a CIAS CSAS or SIAS test signal, each isolation valve actuates to its isolation position.

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

* A CLOSED, ISOLATED, OR BLANK FLANGE VALVE IS CONSIDERED OPERABLE FOR CONTAINMENT ISOLATION

PROOF AND REVIEW

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TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS)	
RD-UV 023	9	Containment radwaste sump pump to LRS holdup tank	~~20~~ 59
RD-UV 024	9	Containment radwaste sump pump to LRS holdup tank	~~8~~ 59
RD-UV 407	9	Containment radwaste sump post-accident sampling system	~~8~~ 59
SG-HV 200	11	Downcomer feedwater chemical injection	~~2~~ 59
SG-HV 201	12	Downcomer feedwater chemical injection	~~2~~ 59
SI-UV 708	23	Containment recirc sump to post-accident sampling system	~~5~~ 59
HC-UV 044	25A	Containment air radioactivity monitor (inlet)	~~2~~ 5
HC-UV 045	25A	Containment air radioactivity monitor (inlet)	~~4~~ 5
HC-UV 046	25B	Containment air radioactivity monitor (outlet)	~~4~~ 5
HC-UV 047	25B	Containment air radioactivity monitor (outlet)	~~2~~ 5
GA-UV 002	29	N_2 to steam generator and reactor drain tank	~~10~~ 59
GA-UV 001	30	N_2 to SI tanks	
~~IA-UV 002~~	~~31~~	~~Service air to reactor containment instrument air~~	~~10~~ ~~59~~
~~NC-UV 401~~	~~33~~	~~NC water to RCP motor bearing lube oil and air coolers~~	~~10~~ ~~59~~
~~NC-UV 403~~	~~34~~	~~NC water to RCP motor bearing lube oil and air coolers~~	~~10~~ 59

PROOF AND REVIEW

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS) (Continued)	
~~NC-UV 402~~	~~34~~	~~NC water to RCP motor bearing lube oil and air coolers~~	~~10~~ 59
HP-UV 001	35	Containment to hydrogen recombiner	~~12~~ 59
HP-UV 003	35	Containment to hydrogen recombiner	~~12~~ 59
HP-UV ~~00~~002	36	Containment to hydrogen recombiner	~~12~~ 59
HP-UV 00~~3~~4	~~38~~ 36	Containment to hydrogen recombiner	~~12~~ 59
HP-UV 005	38	H_2 recombiner return to containment (inlet)	~~12~~ 59
HP-UV 023	38	H_2 control system	~~8~~ 59
HP-UV 006	39	H_2 recombiner return to containment (inlet)	~~12~~ 59
CH-UV 516	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5
CH-UV 523	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5
CH-UV 524	40	Letdown line to post-accident sampling system	5
SS-UV 201	42A	Pressurizer sample surge line	~~5~~ 59
SS-UV 204	42A	Pressurizer sample surge line	~~5~~ 59
SS-UV 202	42B	Pressurizer sample surge line	~~5~~ 59
SS-UV 205	42B	Pressurizer sample surge line	~~5~~ 59
SS-UV 200	42C	Pressurizer sample surge line	~~5~~ 59
SS-UV 203	42C	Pressurizer sample surge line	~~5~~ 59

PROOF AND REVIEW

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS) (Continued)	
~~CH-UV 505~~	~~43~~	~~RC pump seal bleed-off~~	~~5~~
~~CH-UV 506~~	~~43~~	~~RC pump seal bleed-off~~	~~5~~
CH-UV 560	44	Reactor train tank to pre-holdup ion exchanger	5
CH-UV 561	44	Reactor train tank to pre-holdup ion exchanger	5
CH-UV 580	45	Makeup to reactor drain tank	5
CH-UV 715	45	Makeup to reactor drain tank post-accident sampling system	~~5~~ 59
GR-UV 001	52	RDT vent to WG surge tank	~~12~~ 59
GR-UV 002	52	RDT vent to WG surge tank	~~10~~ 59
WC-UV 63	60	Normal chilled water to containment ACU (inlet)	~~10~~ 59
WC-UV 61	61	Normal chilled water to containment ACU (outlet)	~~10~~ 59
WC-UV 62	61	Normal chilled water to containment ACU (outlet)	~~10~~ 59

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		B. CONTAINMENT PURGE (CPIAS)	
CP-UV 002A	56	Containment purge (inlet)	~~10~~ 59
CP-UV 003A	56	Containment purge (inlet)	~~10~~ 59
CP-UV 002B	57	Containment purge (outlet)	~~10~~ 59
CP-UV 003B	57	Containment purge (outlet)	~~10~~ 59
CP-UV 004A	78	Containment purge (inlet)	5
CP-UV 005A	78	Containment purge (inlet)	5
CP-UV 004B	79	Containment purge (outlet)	5
CP-UV 005B	79	Containment purge (outlet)	5

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		C. CONTAINMENT SPRAY (CSAS)	
IA-UV-002	31	Service air to reactor containment inst. air	59
NC-UV-401	33	NC water to RCP motor bearing lube oil and air coolers	59
NC-UV-403	34	NC water to RCP motor bearing lube oil and air coolers	59
NC-UV-402	34	NC water to RCP motor bearing lube oil and air coolers	59
CH-UV-505	43	RC pump seal bleedoff	5
CH-UV-506	43	RC pump seal bleedoff	5

PROOF AND REVIEW

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		C. NORMALLY OPEN - ESF ACTUATED CLOSED	
SG-UV 170	1	Main steam isolation	(a)
SG-UV 171	2	Main steam isolation	(a)
SG-UV 169	1 & 2	Main steam isolation bypass	N.A.
SG-UV 180	3	Main steam isolation	(a)
SG-UV 181	4	Main steam isolation	(a)
SG-UV 183	3 & 4	Main steam isolation bypass	N.A.
SG-UV 1133	1-4	Steam trap/bypass	N.A.
SG-UV 1134	1-4	Steam trap/bypass	N.A.
SG-UV 1135A	1-4	Steam trap/bypass	N.A.
SG-UV 1135B	1-4	Steam trap/bypass	N.A.
SG-UV 1136A	1-4	Steam trap/bypass	N.A.
SG-UV 1136B	1-4	Steam trap/bypass	N.A.
SG-UV 174	8	Steam generator feedwater	N.A.
SB-UV 132	8	Steam generator feedwater	N.A.
SG-UV 137	10	Steam generator feedwater	N.A.
SG-UV 177	10	Steam generator feedwater	N.A.
SG-UV 130	11	Downcomer FIV	N.A.
SG-UV 172	11	Downcomer FIV	N.A.
SG-UV 135	12	Downcomer FIV	N.A.

(a) Tested pursuant to Surveillance Requirement 4.7.1.5.

Delete

PROOF AND REVIEW

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		C. NORMALLY OPEN - ESF ACTUATED CLOSED (Continued)	
SG-UV 175	12	Downcomer FIV	N.A.
SG-UV 682	28	SI drain from drain tank	N.A.
SG-UV 211	37A	Steam generator blowdown sample	N.A.
SG-UV 228	37A	Steam generator blowdown sample	N.A.
SG-UV 204	37B	Steam generator blowdown sample	N.A.
SG-UV 219	37B	Steam generator blowdown sample	N.A.
SG-UV 500P	46	Steam generator blowdown to SCCS	N.A.
SG-UV 500Q	46	Steam generator blowdown to SCCS	N.A.
SG-UV 500R	47	Steam generator blowdown to SCCS	N.A.
SG-UV 500S	47	Steam generator blowdown to SCCS	N.A.
SG-UV 226	48	Steam generator blowdown to downcomer blowdown sample	N.A.
SG-UV 227	48	Steam generator blowdown to downcomer blowdown sample	N.A.
SG-UV 220	49	Steam generator blowdown to downcomer blowdown sample	N.A.
SG-UV 221	49	Steam generator blowdown to downcomer blowdown sample	N.A.
SG-UV 224	63A	SG2 blowdown sample	N.A.
SG-UV 225	63A	SG2 blowdown sample	N.A.
SG-UV 222	63B	SG2 blowdown sample	N.A.
SG-UV 223	63B	SG2 blowdown sample	N.A.

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		D. REQUIRED OPEN DURING ACCIDENT CONDITIONS	
SI-UV 654	26	From shutdown cooling RC loop 2	N.A.
SI-UV 656	26	From shutdown cooling RC loop 2	N.A.
SI-HV 690	26	From shutdown cooling RC loop 2	N.A.
SI-UV 653	27	From shutdown cooling RC loop 1	N.A.
SI-UV 655	27	From shutdown cooling RC loop 1	N.A.
SI-HV 691	27	From shutdown cooling RC loop 1	N.A.
HC-HV 076	32A	Containment pressure monitor	N.A.
HP-HV 007A	35	Containment to hydrogen monitor	N.A.
HP-HV 008A	36	Containment to hydrogen monitor	N.A.
HP-HV 007B	38	Hydrogen monitor to containment	N.A.
HP-HV 008B	39	Hydrogen monitor to containment	N.A.
CH-HV 524	41	Regenerative heat exchanger to RC loop 2A	N.A.
HC-HV 074	54A	Containment pressure monitor	N.A.
HC-HV 075	55A	Containment pressure monitor	N.A.
HC-HV 077	62A	CB pressure monitor	N.A.
SI-HV 331	67	Long-term recirculation loop 2	N.A.
CH-HV 255	72	RC pump seal injection water to RCP 1A, 1B, 2A, 2B	N.A.
SI-HV 321	77	Long-term recirculation loop 1	N.A.
SG-UV 134	2	Main steam to auxiliary feedwater turbine	N.A.

Delete

PROOF AND REVIEW

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		D. REQUIRED OPEN DURING ACCIDENT CONDITIONS (Continued)	
SG-UV 138	3	Main steam to auxiliary feedwater turbine	N.A.
SI-UV 616	13	HPSI to RC loop 2A	N.A.
SI-UV 617	13	HPSI to RC loop 2A	N.A.
SI-UV 626	14	HPSI to RC loop 2B	N.A.
SI-UV 627	14	HPSI to RC loop 2B	N.A.
SI-UV 636	15	HPSI to RC loop 1A	N.A.
SI-UV 637	15	HPSI to RC loop 1A	N.A.
SI-UV 646	16	HPSI to RC loop 1B	N.A.
SI-UV 647	16	HPSI to RC loop 1B	N.A.
SI-UV 615	17	LPSI to RC loop 2A	N.A.
SI-UV 625	18	LPSI to RC loop 2B	N.A.
SI-UV 635	19	LPSI to RC loop 1A	N.A.
SI-UV 645	20	LPSI to RC loop 1B	N.A.
SI-UV 672	21	Shutdown cooling heat exchanger 1 to containment spray header 1	N.A.
SI-UV 671	22	Shutdown cooling heat exchanger 2 to containment spray header 2	N.A.
SI-UV 673	23	Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B	N.A.
SI-UV 674	23	Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B	N.A.

DELETE

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		D. REQUIRED OPEN DURING ACCIDENT CONDITIONS (Continued)	
SI-UV 675	24	Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B	N.A.
SI-UV 676	24	Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B	N.A.
AF-UV 034	75	Steam generator 1 auxiliary feedwater	N.A.
AF-UV 036	75	Steam generator 1 auxiliary feedwater	N.A.
AF-UV 035	76	Steam generator 2 auxiliary feedwater	N.A.
AF-UV 037	76	Steam generator 2 auxiliary feedwater	N.A.

Delete

TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		E. NORMALLY CLOSED/POST ACCIDENT CLOSED	
SG-V-603	1	N_2 blanket supply/N_2 vent	N.A.
SG-V-611	3	N_2 blanket supply/N_2 vent	N.A.
SG-HV 184	1	Main steam atmospheric dump	N.A.
SG-HV 178	2	Main steam atmospheric dump	N.A.
SG-HV 185	3	Main steam atmospheric dump	N.A.
SG-HV 179	4	Main steam atmospheric dump	N.A.
DW-V 061*	6	Containment demineralized water stations	N.A.
DW-V 062*	6	Containment demineralized water stations	N.A.
FP-V 089	7	Fire protection containment	N.A.
SI-V 463*	28	Safety injection drain from drain tank	N.A.
CH-V 854*	41	Chemical addition unit to regenerative heat exchanger	N.A.
PC-V 070	5[illegible]	Fuel pool cooling	N.A.
PC-V 071	50	Fuel pool cooling	N.A.
PC-V 075	51	Refueling pool cleanup	N.A.
PC-V 076	51	Refueling pool cleanup	N.A.
IA-V 072*	59	Containment service air utility station	N.A.

Delete

*May be opened on an intermittent basis under administrative control.

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		F. SAFETY/RELIEF VALVES	
SG-PSV 572	1	Main steam relief	N.A.
SG-PSV 573	1	Main steam relief	N.A.
SG-PSV 574	1	Main steam relief	N.A.
SG-PSV 575	1	Main steam relief	N.A.
SG-PSV 692	1	Main steam relief	N.A.
SG-PSV 576	2	Main steam relief	N.A.
SG-PSV 577	2	Main steam relief	N.A.
SG-PSV 578	2	Main steam relief	N.A.
SG-PSV 579	2	Main steam relief	N.A.
SG-PSV 691	2	Main steam relief	N.A.
SG-PSV 554	3	Main steam relief	N.A.
SG-PSV 555	3	Main steam relief	N.A.
SG-PSV 556	3	Main steam relief	N.A.
SG-PSV 557	3	Main steam relief	N.A.
SG-PSV 695	3	Main steam relief	N.A.
SG-PSV 558	4	Main steam relief	N.A.
SG-PSV 559	4	Main steam relief	N.A.
SG-PSV 560	4	Main steam relief	N.A.
SG-PSV 561	4	Main steam relief	N.A.
SG-PSV 694	4	Main steam relief	N.A.

DELETE

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		F. SAFETY/RELIEF VALVES (Continued)	
SI-PSV 151	23	Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B	N.A.
SI-PSV 140	24	Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B	N.A.
SI-PSV 189	26	From shutdown cooling RC loop 2	N.A.
SI-PSV 179	27	From shutdown cooling RC loop 1	N.A.
SI-PSV 474	28	Safety injection drain relief	N.A.

DELETE

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		G. CHECK VALVES	
FP-V 090	7	Containment fire protection	N.A.
SG-V 003	8	Steam generator feedwater	N.A.
SG-V 007	8	Steam generator feedwater	N.A.
SG-V 005	10	Steam generator feedwater	N.A.
SG-V 006	10	Steam generator feedwater	N.A.
SG-V 642	11	Feedwater downcomer	N.A.
SG-V 652	11	Feedwater downcomer	N.A.
SG-V 653	12	Feedwater downcomer	N.A.
SG-V 693	12	Feedwater downcomer	N.A.
SI-V 113	13	HPSI to RC loop 2A	N.A.
SI-V 123	14	HPSI to RC loop 2B	N.A.
SI-V 133	15	HPSI to RC loop 1A	N.A.
SI-V 143	16	HPSI to RC loop 1B	N.A.
SI-V 114	17	LPSI to RC loop 2A	N.A.
SI-V 124	18	LPSI to RC loop 2B	N.A.
SI-V 134	19	LPSI to RC loop 1A	N.A.
SI-V 144	20	LPSI to RC loop 1B	N.A.
SI-V 164	21	Shutdown cooling heat exchanger 1 to containment spray header 1	N.A.
SI-V 165	22	Shutdown cooling heat exchanger 2 to containment spray header 2	N.A.
GA-V 015	29	N_2 to steam generator and reactor drain tank	N.A.

Delete

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TABLE 3.6-1 (Continued

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		G. CHECK VALVES (Continued)	
GA-V 011	30	N_2 to SI tanks	N.A.
IA-V 021	31	Service air to reactor containment instrument air header	N.A.
NC-V 118	33	NC water to RCP motor bearing lube oil and air coolers	N.A.
HP-V 002	38	H_2 recombiner return to containment	N.A.
HP-V 004	39	H_2 recombiner return to containment	N.A.
CH-VM 70	41	Regenerative heat exchanger to RC loop 2A	N.A.
CH-V 494	45	Makeup to reactor drain tank	N.A.
IA-V 073	59	Containment service air utility station	N.A.
WC-V 039	60	Normal chilled water to containment ACU	N.A.
SI-V 533	67	Long term recirculation loop 2	N.A.
CH-V 835	72	RC pump seal injection water to RCP 1A, 1B, 2A, 2B	N.A.
AF-V 079	75	Steam generator 1 auxiliary feedwater	N.A.
AF-V 080	76	Steam generator 2 auxiliary feedwater	N.A.
SI-V 523	77	Long term recirculation loop 1	N.A.

N.A. - Actuation time not applicable.

Delete

PROOF AND REVIEW

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a) With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

b. WITH 2 HYDROGEN MONITORS INOPERABLE, RESTORE AT LEAST ONE MONITOR WITHIN 48 HOURS OR BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS

c. WITH ONE HYDROGEN MONITOR INOPERABLE, THE PROVISIONS OF SPECIFICATION OF 3.0.4 ARE NOT APPLICABLE

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ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two portable independent containment hydrogen recombiner systems shared among the three units shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or meet the requirements of Specification 3.6.4.3, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by:

   1. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure and control console.

   2. Operating the air blast heat exchanger fan motor and enclosed blower motor continuously for at least 30 minutes.

b. At least once per year by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation.

   2. Performing a "Low-Level Test-Heater Power Off" and "Low-Level Test-Heater Power On" test and verifying that the recombiner temperature increases to and is maintained at 600 ± 25°F for at least one hour. With power off and a simulated input signal of 1280°F, verify the OPERABILITY of all control circuits. When this test is conducted, the air blast heat exchanger fan motor and enclosed blower motor shall be operated continuously for at least 30 minutes.

c. At least once per 5 years by performing a Recombiner System "High-Level Test" and verifying that the recombiner temperature increases to and is maintained at 1200 ± 50°F for at least one hour.

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HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus. ~~in each of the three units~~.

APPLICABILITY: MODES 1* and 2.*

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

*With less than two hydrogen recombiners OPERABLE.

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3. Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

PRE FILTERS

1. Verifying that the pressure drop across the combined HEPA filters, and charcoal adsorber banks is less than ~~(8)~~ 8.4 inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%.

2. Verifying that the heaters dissipate AT LEAST 0.5 ~~± 0.1~~ kW when tested in accordance with ANSI N510-~~1975.)~~ 1980

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam line code safety valves inoperable per steam generator, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam line code safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER S/G No. 1	S/G No. 2	LIFT SETTING (±1%)*	MINIMUM RATED CAPACITY**
a. PSV 572	PSV 554	1250 psig	941,543 lb/hr
b. PSV 579	PSV 561	1250 psig	941,543 lb/hr
c. PSV 573	PSV 555	1290 psig	971,332 lb/hr
d. PSV 578	PSV 560	1290 psig	971,332 lb/hr
e. PSV 574	PSV 556	1315 psig	989,950 lb/hr
f. PSV 575	PSV 557	1315 psig	989,950 lb/hr
g. PSV 576	PSV 558	1315 psig	989,950 lb/hr
h. PSV 577	PSV 559	1315 psig	989,950 lb/hr
i. PSV 691	PSV 694	1315 psig	989,950 lb/hr
j. PSV 692	PSV 695	1315 psig	989,950 lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation. These capacities provide a minimum total capacity of 19,530,900 lb/hr at 1355 psig (1315 psig + 3% accumulation).

PROOF AND REVIEW

STEADY STATE POWER LEVEL And T[illegible] 3.7-2

MAXIMUM ALLOWABLE VARIABLE OVERPOWER HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING TWO LOOP OPERATION WITH FOUR PUMPS OPERATING

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)	MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (PERCENT OF RATED THERMAL POWER)
1	108.0	98.2
2	97.1	87.3
3	86.2	76.4
4	75.3	65.5

PROOF AND REVIEW

PROOF AND REVIEW

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LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each AFW pump, and the manual discharge line valve of each AFW pump are locked, ~~in the open position~~ SEALED, OR OTHERWISE SECURED IN THE OPEN POSITION.

*Until the steam generators are no longer required for heat removal.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued).

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

CAPS

REFUELING SHUTDOWN OR THAT

c. Prior to startup following any cold shutdown of 30 days or longer, by verifying (by means of a flow test) the normal flow path from the condensate storage tank to each of the steam generators through each of the auxiliary feedwater pumps. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump. OR MODE 4

d.

DELIVERS 750 gpm AT 1250 psig OR EQUIVALENT

PROOF AND REVIEW

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 23 feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3#, and 4*.#

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank to auxiliary feed system isolation valves are open, and

b. That the reactor makeup water tank contains water levels of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

NOT APPLICABLE WHEN COOLDOWN IS IN PROGRESS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

PROOF AND REVIEW

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	(a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. (b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1: IN MODE 2

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least ~~HOT STANDBY~~ within the next 6 hours, ~~and in COLD SHUTDOWN within the following 30 hours~~.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 4 to perform the surveillance testing of Specification 4.7.1.5.1 provided the testing is performed within 12 hours after achieving NORMAL ~~sufficient~~ steam pressure to perform the test.
OPERATING AND NORMAL OPERATING temperature FOR THE SECONDARY SIDE
MODE 3 OR

PLANT SYSTEMS

ATMOSPHERE DUMP VALVES

LIMITING CONDITIONS FOR OPERATIONS

3.7.1.6 The atmospheric dump valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*

ACTION:

With less than one atmospheric dump valve per steam generator OPERABLE, restore the required atmospheric dump valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying nitrogen accumulator tank at a pressure $\geq$ 400 PSIG.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer verify that all valves will open and close fully.

* When steam generators are being used for decay heat removal.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 120°F when the pressure of the secondary coolant in the steam generator is greater than 230 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 230 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in the secondary side of the steam generators shall be determined to be less than 230 psig at least once per hour when the temperature of the secondary coolant is less than 120°F.

SHIFT

PROOF AND REVIEW

3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent essential cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one essential cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two essential cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

c. At least once per 18 months during shutdown, by verifying that the essential cooling water pumps start on an SIAS test signal.

3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent essential spray pond loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one essential spray pond loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two essential spray pond loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with two essential spray ponds each with:

a. A minimum water depth of 12 feet, and

b. An average water temperature of less than or equal to 105°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water depth to be within their limits for each essential spray pond.

PROOF AND REVIEW

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 At least two independent essential chilled water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

7 DAYS

a) With only one essential chilled water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6 At least two essential chilled water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b) WITH ONLY ONE EMERGENCY CHILLED WATER SYSTEM OPERABLE:

1. WITHIN 1 HOUR VERIFY THAT THE NORMAL HVAC SYSTEM IS PROVIDING SPACE COOLING TO THE VITAL POWER DISTRIBUTION ROOMS THAT DEPEND ON THE INOPERABLE ESSENTIAL CHILLED WATER SYSTEM FOR SPACE COOLING, AND
2. WITHIN 8 HOURS ESTABLISH OPERABILITY OF THE SAFE SHUTDOWN SYSTEMS WHICH DO NOT DEPEND ON THE INOPERABLE ESSENTIAL CHILLED WATER SYSTEM (ONE TRAIN EACH OF BORATION, PRESSURIZER HEATERS AND AUXILIARY FEEDWATER)
3. WITHIN 24 HOURS ESTABLISH OPERABILITY OF ALL REQUIRED SYSTEMS, SUBSYSTEMS, TRAINS, COMPONENTS AND DEVICES THAT DEPEND ON THE REMAINING OPERABILITY ESSENTIAL CHILLED WATER SYSTEM FOR SPACE COOLING

IF THESE CONDITIONS ARE NOT SATISFIED WITHIN THE SPECIFIED TIME, BE IN AT LEAST HOT STANDBY WITHIN THE NEXT 6 HOURS IN COLD SHUTDOWN WITHIN THE FOLLOWING 30 HOURS.

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3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room essential filtration systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room essential filtration system in the recirculation mode.

b. With both control room essential filtration systems inoperable, or with the OPERABLE control room essential filtration system, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room essential filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

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SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a control room essential filtration actuation, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation.

1/8

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

PROOF AND REVIEW

3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ESF pump room air exhaust cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ESF pump room air exhaust cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 Each ESF pump room air exhaust cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm $\pm$ 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 6000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm $\pm$ 10%.

2. Verifying that the system starts on an SIAS test signal.

e. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm $\pm$ 10%.

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

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No. of Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

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e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested.

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon

as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

PROOF AND REVIEW

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers OR

Snubbers which fail the visual inspection of the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit. THESE

FUNCTIONAL

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

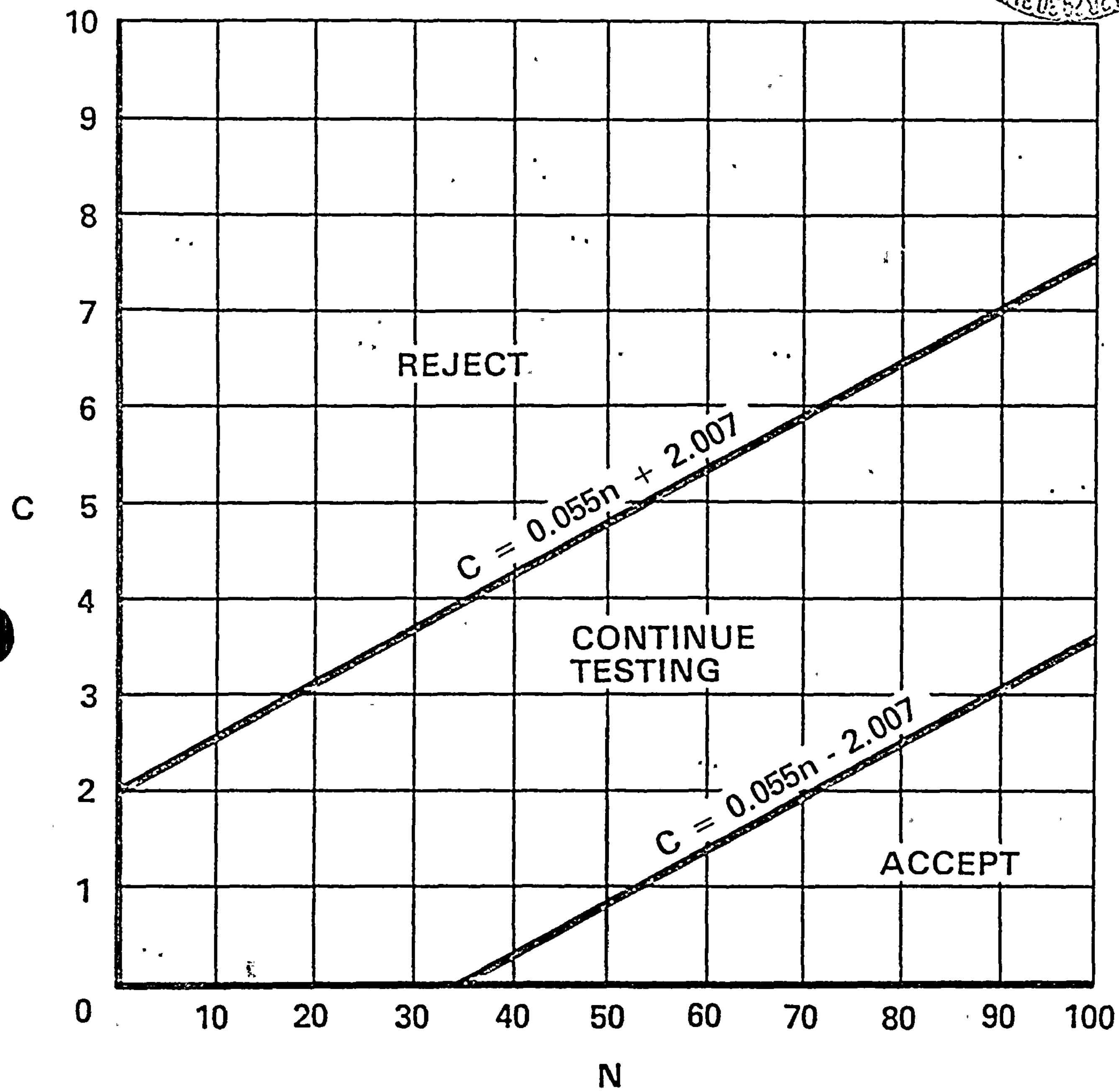

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

  1. Decontaminate and repair the sealed source, or

  2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test m.chod shall have a detection sensitivity of at least 0.005 microcurie per test sample.

4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material:

  1. With a half-life greater than 30 days (excluding Hydrogen 3), and

  2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcurie of removable contamination.

PROOF AND REVIEW

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. Three 50% capacity fire suppression pumps, each with a capacity of 1500 gpm ± 10% with their discharge aligned to the fire suppression header,

b. Two separate water supply tanks, each with a minimum contained volume of 300,000 gallons (23 FEET 1 1/2 INCHES), and

c. An OPERABLE flow path capable of taking suction from the T01-A tank and the T01-B tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.5, and 3.7.11.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position, WHEN REQUIRED TO BE OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each pump develops ~~at least~~ 1500 gpm $\pm 10\%$ at a system head of 125 psig, BY RECORDING READINGS FOR AT LEAST THREE POINTS ON THE TEST CURVE
2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and
3. Verifying that each fire suppression pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 85 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.11.1.2 The fire pump diesel engines shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by verifying:

1. The diesel fuel oil day storage tanks each contain at least 315 gallons of fuel, and
2. The diesel engines start from ambient conditions and operate for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.1.1.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

   1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

PROOF AND REVIEW

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems, listed in Table 3.7-3, shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal/smoke test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

PROOF AND REVIEW

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

a. Lower Cable Spreading Room Zone 14 - Control Building 120 ft Elevation

b. Upper Cable Spreading Room Zone 20 - Control Building 160 ft Elevation

c. Diesel Generator Room, one Train A, one Train B Zone 21 - Diesel Generator Building 100 ft Elevation

d. Fuel Oil Day Tank Vaults, one Train A, one Train B Zone 23 - Diesel Generator Building 131 ft Elevation

e. Containment Spray Pumps Room, one Train A, one Train B Zone 30 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

f. High Pressure Safety Injection Pump Rooms, one Train A, one Train B Zone 31 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

g. Low Pressure Safety Injection Pump Rooms, one Train A, one Train B Zone 32 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation

h. Electrical Penetration Room, one Train A (Channel C) Zone 42A - Auxiliary Building 100 ft Elevation

i. Electrical Penetration Room, one Train B (Channel B) Zone 42B - Auxiliary Building 100 ft Elevation

j. Corridors Zone 42C - Auxiliary Building 100 ft Elevation

k. Corridors Zone 42D - Auxiliary Building 100 ft Elevation

l. Electrical Penetration Rooms, one Train A (Channel A) Zone 47A - Auxiliary Building 120 ft Elevation

m. Electrical Penetration Rooms, one Train A (Channel D) Zone 47B - Auxiliary Building 120 ft Elevation

n. Central Corridors Zone 52A - Auxiliary Building 120 ft Elevation

o. Central Corridors Zone 52D - Auxiliary Building 120 ft Elevation

p. Turbine-Driven Auxiliary Feed Pump Room Zone 72 - Main Steam Support Structure 81

q. ESF Transformers and 13.8 kV Switchgear - each transformer Zone 76 - Outside Areas 100 ft Elevation

r. Compartments between Auxiliary & Control Buildings between 74 ft & 156 ft 4 inch Elevation on Trains A & B Zone 86

New TABLE Coming

PROOF AND REVIEW

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

1. Lower Cable Spreading Room Zone 14 - Control Building 120 ft Elevation.

a.	System	1.
b.	System	2.
c.	System	3.
d.	System	4.
e.	System	5.
f.	System	6.

2. Upper Cable Spreading Room Zone 20 - Control Building 160 ft Elevation.

3. Diesel Generator Room, Train A, Zone 21 - Diesel Generator Building 100 ft Elevation.

4. Diesel Generator Room, Train B, Zone 21 - Diesel Generator Building 100 ft Elevation.

5. Fuel Oil Day Tank Vault, Train A, Zone 23 - Diesel Generator Building 131 ft Elevation.

6. Fuel Oil Day Tank Vault, Train B, Zone 23 - Diesel Generator Building 131 ft Elevation.

7. Low Pressure Safety Injection Pump Room, Train B, Zone 32 - Auxiliary Building 40 ft & 51 ft 6 inch Elevation.

8. Electrical Penetration Room, Train A (Channel C) Zone 42A - Auxiliary Building 100 ft Elevation.

9. Electrical Penetration Room, Train B (Channel B) Zone 42B - Auxiliary Building 100 ft Elevation.

10. Charging Pumps A, B and E Zones 46. Corridors Zone 42C Auxiliary Building 100 ft Elevation.

11. Corridors Zone 42D - Auxiliary Building 100 ft Elevation.

12. Electrical Penetration Room, Train A (Channel A) Zone 47A - Auxiliary Building 120 ft Elevation.

13. Electrical Penetration Room, Train A (Channel D) Zone 47B - Auxiliary Building 120 ft Elevation.

14. Central Corridors Zone 52A - Auxiliary Building 120 ft Elevation.

15. Central Corridors Zone 52D - Auxiliary Building 120 ft Elevation.

16. Turbine - Driven Auxiliary Feed Pump Room Zone 72 - Main Steam Support Structure 81.

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

17. Compartments between Auxiliary & Control Buildings between 74 ft & 156 ft 4 inch Elevation on Train A, Zone 86.

18. Compartments between Auxiliary & Control Buildings 74 ft & 156 ft 4 inch Elevation on Train B, Zone 86.

19. Main Steam Support Structure 100 ft through 140 ft Elevation.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following low pressure CO_2 systems shall be OPERABLE.

a. ESF Switchgear Room; one Train A, one Train B Zone 5 Control Building 100 ft Elevation

b. Battery Rooms; one Train A (Channel C) one Train B (Channel D) Zone 8 Control Building 100 ft Elevation

c. Battery Rooms; one Train A (Channel A) one Train B (Channel B) Zone 9 Control Building 100 ft Elevation

APPLICABILITY: Whenever equipment protected by the CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.11.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank weight to be greater than 10000 lb and pressure to be greater than 275 psig, and

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying:

1. The system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."
BY VISUAL INSPECTION TO VERIFY NO OBSTRUCTIONS IN THE DISCHARGE PATH OF THE NOZZLES OR BY "PUFF TEST."

PROOF AND REVIEW

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.5 The fire hose stations shown in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE, except that fire hose stations located in containment shall have their containment isolation valves closed in MODES 1, 2, 3, 4*, and 5*.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-4 inoperable, route a fire hose to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself,

   a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose and related valves shall be mounted above the hose and at the inoperable hose station.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.
   2. Removing the hose for inspection and reracking, and
   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

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*If maintenance is to be performed in containment during MODE 4 or 5, the fire hose stations located in containment shall have their containment isolation valves open during the period the maintenance is being performed.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.
2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

PROOF AND REVIEW

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK IDENTIFICATION
Containment SW	80'	HS #03
Containment NW	80'	HS #04
Containment NE	100'	HS #05
Containment SE	100'	HS #06
Containment SW	100'	HS #07
Containment NW	100'	HS #08
Containment NE	120'	HS #09
Containment SE	120'	HS #10
Containment SW	120'	HS #11
Containment NW	120'	HS #12
Auxiliary Bldg. North Corridor - W	40'	HS #17
Auxiliary Bldg. North Corridor - E	40'	HS #18
Auxiliary Bldg. North Corridor - W	51'6"	HS #21
Auxiliary Bldg. North Corridor - E	51'6"	HS #22
Auxiliary Bldg. SE	70'	HS #23
Auxiliary Bldg. SW	70'	HS #24
Auxiliary Bldg. NW	70'	HS #25
Auxiliary Bldg. North Center Corridor	70'	HS #26
Auxiliary Bldg. NE	70'	HS #27
Auxiliary Bldg. NW	88'	HS #30
Auxiliary Bldg. NE	88'	HS #31
Auxiliary Bldg. SW	100'	HS #33
Auxiliary Bldg. East Corridor	120'	HS #37
Auxiliary Bldg. SW	120'	HS #38
Control Bldg. SW	74'	HS #86
Control Bldg. E	74'	HS #87
Control Bldg. SW	100'	HS #88
Control Bldg. East by Elevator	100'	HS #89
Control Bldg. SW	120'	HS #90
Control Bldg. SW	140'	HS #92
Control Bldg. SW	160'	HS #94
Control Bldg. SE	100'	HS #108
Fuel Bldg. South	100'	HS #97
CONTAINMENT NE	80	HS 1
CONTAINMENT SE	80	HS 2
CONTAINMENT NE	140	HS 13
CONTAINMENT SW	140	HS 14

PROOF AND REVIEW

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2-1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged. FOR DAMAGE

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

3. Performing a flow check of each hydrant to verify its OPERABILITY.

PROOF AND REVIEW

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER
150' Plant North of Fuel Bldg.	F. H. #7
100' Plant West of Rad Waste Bldg.	F. H. #9
150' Plant Northwest of Fuel Bldg.	F. H. #8

PROOF AND REVIEW

3/4.7.12 FIRE-RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire-rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: When the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire-rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire-rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

~~b~~ A. That each locked-closed fire door is closed at least once per 7 days.

~~c~~ B. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, ~~and~~ perform~~ing~~ a functional test at least once per 18 months.

c.
d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

PROOF AND REVIEW

3/4.7.13 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, 3, and 4. e

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.13 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot lets, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than ~~370~~ psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than ~~450~~ psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia. 410 500

PROOF AND REVIEW

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Separate day fuel tanks with a minimum level of 2.75 feet (550 gallons of fuel), and

   2. A separate fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4 within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, 3, or 4*, the steam-driven auxiliary feed pump is OPERABLE.

   If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Until the steam generator is no longer required for heat removal.

a. With an offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a diesel generator of the above required A.C. electrical power sources inoperable,* demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours; restore diesel generators to OPERABLE status within (A**) days*** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. At the number of failures for the inoperable diesel indicated in Table 4.8-2 perform the Additional Reliability Actions prescribed in Table 4.8-2 and its attachments.

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter and Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With the diesel generator restored to OPERABLE status, follow Action Statement a. With the offsite circuit restored to OPERABLE status, follow Action Statement b.

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirement 4.8.1.1.2.4 ELECTRIC POWER SYSTEMS

**The maximum time that an individual diesel generator may be inoperable (A) shall be established by the licensee based on the manufacturer's recommendations and previous maintenance and repair experience. Every reasonable effort shall be made to restore individual diesel generators to operable status within that time period (A). Every reasonable effort shall be interpreted to mean that diagnosis and repairs are to begin immediately and are to continue uninterrupted until the diesel generator is declared operable or an orderly retreat to cold shutdown is initiated.

***The maximum total cumulative time that the diesel generators of the onsite emergency AC power system may be in the INOPERABLE status in a given year shall be proposed by the licensee.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 5500 kW in less than or equal to 120 seconds, and operates with a load greater than or equal to 5500 kW for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks.

c. At least once per 92 days and from new fuel prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:

1. A water and sediment content of less than or equal to 0.05 volume percent;

2. A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

3. A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.80 but less than or equal to 0.99 or an API gravity at 60°F of greater than or equal to 11 degrees but less than or equal to 47 degrees;

4. An impurity level of less than 2 mg of insolubles per 100 mL when tested in accordance with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and

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5. The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a SINGLE LARGEST load of greater than or equal to ~~885~~ 839 kW (Train B AFW pump) OR 696 KW (TRAIN A HPSI pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 + 1.2 Hz.

3. Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 6240 volts during and following the load rejection.

4. Simulating a loss-of-offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential, and low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6050 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 5500 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2d.6.b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 5500 kW.

9. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status. (NOT Running)

10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

(Running unloaded)

SURVEILLANCE REQUIREMENTS (Continued)

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± ~~10%~~ 1 second of its design interval.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) (turning gear engaged)
    b) (emergency stop)

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 600 rpm (steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz) in less than or equal to 10 seconds.

f. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or the equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission within 30 days in a Special Report pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last ~~100~~ 20 Valid Tests.*	Test Frequency
≤ 1	At least once per 31 days
~~2~~	~~At least once per 14 days~~
≥2 ~~3~~	At least once per 7 days **
~~≥ 4~~	~~At least once per 3 days~~

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the ~~last 100 tests are~~ NUMBER OF AND FAILURES determined on a per ~~nuclear unit~~ DIESEL GENERATOR basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last ~~100~~ 20 valid tests". ~~Entry into this test schedule shall be made at the 31 day test frequency.~~

** THIS TEST FREQUENCY SHALL BE MAINTAINED UNTIL SEVEN CONSECUTIVE FAILURE FREE DEMANDS HAVE BEEN PERFORMED AND THE NUMBER OF FAILURES IN THE LAST 20 VALID DEMANDS HAS BEEN REDUCED TO ONE OR LESS

Justify by generic NRC letter 84-15

ACTION: (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, follow Action Statement a.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With one diesel generator unit restored, follow Action Statement b and d.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day tank with a minimum level of 2.75 feet (550 gallons of fuel),

   2. A fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for Requirement 4.8.1.1.2a.5.) and 4.8.1.1.3.

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3/4.8.2 D.C. SOURCES

OPERATNG

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the D.C. trains listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required D.C. trains inoperable, restore the inoperable D.C. trains to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required chargers inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

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b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 400 amperes for batteries A and B and 300 amperes for batteries C and D at 125 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 3.8-1

D.C. ELECTRICAL SOURCES

Train A

CHANNEL A	CHANNEL C
125V bus E-PKA-M41	125V D.C. bus E-PKC-M43
125V D.C. battery bank E-PKA-F11	125V D.C. battery bank E-PKC-F13
Battery charger E-PKA-H11	Battery charger E-PKC-H13
or	or
Backup battery charger E-PKA-H15 (AC)	Backup battery charger E-PKA-H15 (AC)

Train B

CHANNEL B	CHANNEL D
125V D.C. bus E-PKB-M42	125V D.C. bus E-PKD-M44
125V D.C. battery bank E-PKB-F12	125V D.C. battery bank E-PKD-F14
Battery charger E-PKB-H12	Battery charger E-PKD-H14
or	or
Backup battery charger E-PKB-H16 (BD)	Backup battery charger E-PKB-H16 (BD)

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(a)	> 2.07 volts
Specific Gravity(b)		$\geq$ 1.195	Not more than 0.020 below the average of all connected cells
	$\geq$ 1.205(c)	Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(c)

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value, declare the battery inoperable.

(a) Corrected for average electrolyte temperature.

(b) Corrected for electrolyte temperature and level.

(c) Or battery charging current is less than 2 amps when on charge.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one D.C. train as listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With a required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required D.C. train to OPERABLE status as soon as possible.

b. With a required charger inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open both between redundant busses within the unit (and between units at the same station).

a. Train "A" A.C. emergency busses consisting of:

   1. 4160-volt ESF Bus #E-PBA-S03
   2. 480-volt ESF Load Center #E-PGA-L31
      a. MCC E-PHA-M31
   3. 480-volt ESF Load Center #EPGA-L33
      a. MCC E-PHA-M33
      b. MCC E-PHA-M37
   4. 480-volt ESF Load Center #E-PGA-L35
      a. MCC E-PHA-M35

b. Train "B" A.C. emergency busses consisting of:

   1. 4160-volt ESF Bus #E-PBB-S04
   2. 480-volt ESF Load Center #E-PGB-L32
      a. MCC E-PHB-M32
      b. MCC E-PHB-M38
   3. 480-volt ESF Load Center #E-PGB-L34
      a. MCC E-PHB-M34
   4. 480-volt ESF Load Center #E-PGB-L36
      a. MCC E-PHB-M36

c. 120-volt Channel A Vital A.C. Bus #E-PNA-D25 energized from its associated inverter connected to D.C. Channel A.*

d. 120-volt Channel B Vital A.C. Bus #E-PNB-D26 energized from its associated inverter connected to D.C. Channel B.*

e. 120-volt Channel C Vital A.C. Bus #E-PNC-D27 energized from its associated inverter connected to D.C. Channel C.*

f. 120-volt Channel D Vital A.C. Bus #E-PND-D28 energized from its associated inverter connected to D.C. Channel D.*

g. 125-volt D.C. Channel A energized from Battery Bank E-PKA-F11.

h. 125-volt D.C. Channel B energized from Battery Bank E-PKB-F12.

i. 125-volt D.C. Channel C energized from Battery Bank E-PKC-F13.

j. 125-volt D.C. Channel D energized from Battery Bank E-PKD-F14.

*Two inverters may be disconnected from their D.C. bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

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ELECTRICAL POWER SYSTEMS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ESF Load Centers not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160-volt A.C. ESF bus, and three 480-volt A.C. load centers and their associated four class 1E-MCCs.

b. Two 120-volt A.C. channel vital busses energized from their associated inverters connected to their respective D.C. channels.

c. One 125-volt D.C. train with both required channels energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective devices shown in Table 3.8-2 inoperable:

a. Restore the protection device(s) to OPERABLE status or deenergize the circuits(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

   (a) A CHANNEL CALIBRATION of the associated protection relays, and

   (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-2.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers' nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets it manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all ´.ses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

PROOF AND REVIEW

TABLE 3.8-2

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1006	E-NHN-M1002B	SG WET LAYUP RECIRC. PUMP M-SGN-P01B
E-NHN-M1017	E-NHN-M1002B	CTMT/RADWASTE SUMP PUMP M-RDN-P03
E-NHN-M1003	E-NHN-M1002A	RCP 1B CONTROLLED BLEEDOFF VLV J-RCE-HV-431
E-NHN-M1004	E-NHN-M1002A	RCP 1B HP COOLER INLET VLV J-RCN-HV-447
E-NHN-M1005	E-NHN-M1002A	RCP 1B HP COOLER OUTLET VLV J-RCN-HV-451
E-NHN-M1010	E-NHN-M1002A	REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-M02B
E-NHN-M1014	E-NHN-M1002A	REACTOR CAVITY SUMP PUMP M-RDN-P01A
E-NHN-M2808	E-NHN-M2832C	RCP 2B CONTROL BLEEDOFF VLV J-RCE-HV-433
E-NHN-M2813	E-NHN-M2832C	RCP 2B HI PRESSURE COOLER INLET VLV J-RCN-HV-449
E-NHN-M1009	E-NHN-M1002A	RCP 2B HI PRESSURE COOLER OUTLET VLV J-RCN-HV-453
E-NHN-M1306	E-NHN-M1314A	SG 2 HOT LEG BLDWN ISO VLV J-SGE-HV-42
E-NHN-M1307	E-NHN-M1314A	SG 2 COLD LEG BLDWN ISO VLV J-SGE-HV-44
E-NHN-M1311	E-NHN-M1314D	WET LAY UP RECIRC PUMP M-SGN-P01A
E-NHN-M1316	E-NHN-M1314C	RCPT (30A) FOR SEAL CRANE ASSY MOTOR E-NHN-122A; E-NHN-122B
E-NHN-M1339	E-NHN-M1314C	MOVABLE INCORE DETECTOR DRIVE MACHINE M-RIN-M03A

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1321	E-NHN-M1344B	WELDING RCPT'S E-NHN-I107A B, C, D
E-NHN-M1331	E-NHN-M1314B	REACTOR CAVITY SUMP PUMP M-RDN-P01B
E-NHN-M1341	E-NHN-M1314B	REACTOR CAVITY FAN C DISCH DAMPER M-HCN-M02C
E-NHN-M1342	E-NHN-M1314B	CEDM ACU A INTAKE DAMPER M-HCN-M03A
E-NHN-M1343	E-NHN-M1314B	CEDM ACU B INTAKE DAMPER M-HCN-M03B
E-NHN-M1323	E-NHN-M1344A	REACTOR COOLANT OIL LIFT PUMP 2A M-RCN-P02C
E-NHN-M1332	E-NHN-M1344A	CTMT RADWASTE SUMP EAST M-RDN-P02
E-NHN-M1503	E-NHN-M1502A	RCP 1A CONTROL BLEEDOFF VLV J-RCE-HV-430
E-NHN-M1504	E-NHN-M1502A	RCP 2A CONTROL BLEEDOFF VLV J-RCE-HV-432
E-NHN-M1505	E-NHN-M1502A	RCP 1A HI PRESSURE COOLER INLET VLV J-RCN-HV-446
E-NHN-M1506	E-NHN-M1502A	RCP 2A HI PRESSURE COOLER INLET VLV J-RCN-HV-448
E-NHN-M1507	E-NHN-M1502A	RCP 1A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-450
E-NHN-M1511	E-NHN-M1535A	WELDING RCPT'S E-NHN-I12A, B, C
E-NHN-M1508	E-NHN-M1502B	RCP 2A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-452
E-NHN-M1509	E-NHN-M1502B	REACTOR CAVITY FAN A DISCH DAMPER M-HCN-M02A

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M1533	E-NHN-M1502B	REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02D
E-NHN-M1534	E-NHN-M1535	CTMT BLDG MONO HOIST 1 TON M-ZCN-009
E-NHN-M1517	E-NHN-M1535	REACTOR COOLANT OIL LIFT PUMP M-RCN-P02A
E-NHN-M1902	E-NHN-M1917A	REACTOR CAVITY NORM CLG FAN M-HCN-A03A
E-NHN-M1904	E-NHN-M1917B	REACTOR CAVITY NORM CLG FAN M-HCN-A03C
E-NHN-M1907	E-NHN-M1917	CEDM NORM ACU-A HEXCH OUTLET VLV J-NCN-HV-485
E-NHN-M1911	E-NHN-M1917	CTMT NORM ACU-C CHILLED WTR INLET VLV J-WCN-HV-59
E-NHN-M1912	E-NHN-M1917	CTMT NORM ACU-A CHILLED WTR INLET VLV J-WCN-HV-57
E-NHN-M2008	E-NHN-M2010	CEDM NORM ACU-B HEXCH OUTLET VLV J-NCN-HV-486
E-NHN-M2003	E-NHN-M2010	CTMT NORM ACU-B CHILL WATER INLET VLV J-WCN-HV-58
E-NHN-M2004	E-NHN-M2010	CTMT NORM ACU-D CHILL WATER INLET VLV J-WCN-HV-60
E-NHN-M2006	E-NHN-M2010A	REACTOR CAVITY NORM CLG FAN M-HCN-A03B
E-NHN-M2007	E-NHN-M2016	REACTOR CAVITY NORM CLG FAN M-HCN-A03D
E-NHN-M2803	E-NHN-M2827A	CEDM ACU C INTAKE DAMPER M-HCN-M03C
E-NHN-M2804	E-NHN-M2827A	CEDM ACU D INTAKE DAMPER M-HCN-M03D

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M2805	E-NHN-M2827A	SG1 COLD LEG BLOWDOWN ISO VLV J-SGE-HV-91
E-NHN-M2806	E-NHN-M2827~~A~~ B	SG HOT LEG BLOWDOWN ISOLATION VALVE J-SGE-HV43
E-NHN-M2827	E-NHN-M2827~~A~~ ~~B~~ A	REACTOR COOL PUMP OIL LIFT PUMP 1B M-RCN-P02B
E-NHN-M2828	E-NHN-M2827~~B~~ A	REACTOR COOLANT PUMP OIL LIFT PUMP 2B M-RCN-P~~2~~02D
E-NHN-M2809	E-NHN-M2827C	CONTAINMENT EQUIP HATCH J-ZCN-E02
E-NHN-M2811	E-NHN-M2832A	30A RECEPTACLES FOR CTMT BLDG JIB CRANE
E-NHN-M2818	E-NHN-M2832A	30A RECEPTACLES FOR SEAL CRANE ASSY MOT
E-NHN-M2817	E-NHN-M2832B	CTMT BLDG MONORAIL HOIST 1 TON M-ZCN-G03
E-NHN-M2819	E-NHN-M2832B	30A RECEPTACLES FOR CTMT BLDG JIB CRANE G04 A, B
E-NHN-M2820	E-NHN-M2832D	CTMT BLDG ELEV #2 CONTROLLER J-ZCN-E01
E-NHN-M2821	E-NHN-M2828C	MULTIPLE STUD TENSIONER M-ZCN-M15
E-NHN-M2822	E-NHN-M2828B	WELDING RECPTS E-NHN-I09 B, C, D
E-NHN-M2801A	E-NHN-M2827B	FUEL TRANSFER SYS CONTROL CONSOLE E-PCE-D02
E-NHN-M2833	E-NHN-M2827B	REFUELING MACHINE E-PCE-J02
E-NHN-M2833A	E-NHN-M2827B	CEA CHANGE PLATFORM E-PCE-J01

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

LEAVE SPACE ACU A

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M7102	E-NHN-M7104	CONTAINMENT NORMAL ACUA DISCHARGE DAMPER M-HCN-M01A
E-NHN-M7103	E-NHN-M7104	CONTAINMENT NORMAL ACUC DISCHARGE DAMPER M-HCN-M01C (LEAVE SPACE ACU C) M01C
E-NHN-M7114	E-NHN-7113	PZR NORMAL COOLING FAN M-HCN-A06A
E-NHN-M2816	E-NHN-M2832C	CTMT BLDG MONORAIL HOIST-2 TON M-ZCN-G08
E-NHN-M2834A	E-NHN-M2832C	MOVABLE INCORE DETECTOR DRIVE MACH #2 M-RIN-M03B
E-NHN-M7202	E-NHN-M7204	CTM NORM ACU B DISCH DAMPER M-HCN-M01B
E-NHN-M7203	E-NHN-M7204	CTM NORM ACU D DISCH DAMPER M-HCN-M01D
E-NHN-M7214	E-NHN-M7213	PZR NORMAL COOLING FAN M-HCN-A06B
E-PGA-L31E2	E-NGN-B31E2 (FUSE)	CONTAINMENT NORMAL ACU FAN M-HCN-A01A
E-PGA-L31E3	E-NGN-B31E3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02A
E-PGB-L32E3	E-NGN-B32E3 (FUSE)	PRESSURIZER BACKUP HEATERS M-RCE-B1~~6~~ 8, B10, A5
E-PGB-L32E2	E-NGN-B32E2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02B
E-PGA-L33D2	E-NGN-B33D2 (FUSE)	CONTAINMENT NORMAL ACU FAN M-HCN-A01C
E-PGA-L33D4	E-NGN-B33D4 (FUSE)	PRESSURIZER BACKUP HTR, M-RCE B1, B9, A14
E-PGA-L33D3	E-NGN-B33D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02C

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PGB-L34D2	E-NGN-B34D2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A01D
E-PGB-L34D3	E-NGN-B34D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02D
E-PGB-L36D3	E-NGN-~~B63D3~~ B36D3 (FUSE)	CTMT NOR ACU FAN M-HCN-A01B
E-PHA-M3318	E-PHA-M3334	SAFETY INJECT TANK 4 ISOL VLV J-SIA-UV-644
E-PHA-M3316	E-PHA-M3316A	SAFETY INJECT TANK 3 ISOL VLV J-SIA-UV-634
E-PHB-M3404	E-PHB-M3405B	NCWS RET INT CTMT ISOL VLV J-NCB-UV-403
E-PHA-M3519	E-PHA-M3521A	CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-48
E-PHA-M3521	E-PHA-M3517	CTMT PRG RFL MODE ISO VLV J-CPA-UV-2B
E-PHA-M3503	E-PHA-M3507A	SHUT DN CLG ISOL LOOP 1 VLV J-SIA-UV-651
E-PHA-M3508	E-PHA-M3511A	CTMT/RAD SUMP CTMT INT ISO VLV J-RDA-UV-23
E-PHA-M3512	E-PHA-M3513A	CTMT SUMP ISOL TRAIN A VLV J-SIA-UV-673
E-PHB-M3622	E-PHB-M3629	CTMT PRG REFULING MODE ISO VLV J-CPB-UV-3A
E-PHB-M3604	E-PHB-M3604A	SHUT DN CLG ISOL LOOP 2 VLV J-SIB-UV-652
E-PHB-M3619	E-PHB-M3641A	SAFETY INJECTION TANK ISOL VLV J-SIB-UV-614
E-PHB-M3624	E-PHA-M3607A	CTMT PRG PWR ACCESS MODE ISO VLV J-CPB-UV-5A

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PHB-M3613	E-PHB-M3613A	CTMT SUMP ISOL TRAIN B VLV J-SIB-UV-675
E-PHB-M3618	E-PHB-M3641	SAFETY INJECTION TANK 2 ISO VLV J-SIB-UV-624
E-PHA-M3704	E-PHA-M3703A	WASTE GAS HEATER CONT. ISOLATION VALVE GRA-UV1
E-PHA-M3715	E-PHA-M3719	H_2 CONT TRAIN A UPSTM SUP ISO VLV J-HPA-UV-1
E-PHB-M3816	E-PHB-M3836	H_2 CTMT TRAIN B UPSTM SUP ISO VLV J-HPB-UV-2
E-PHB-M3811	E-PHB-M3813A	NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61
E-PKD-B44	E-PKD-M4411	SHUTDOWN CLG ISOL VLV J-SID-UV-654
E-PKC-B43	E-PKC-M4311	SHUTDOWN COOLING ISOL VLV J-SIC-UV-653
E-NNN-D1113	E-NNN-D11	MOVABLE INCORE DRIVE SYS #I 800VA, M-RIN-M03A VIA E-RIN-J01A
E-NNN-D1213	E-NNN-D12	MOVABLE INCORE DRIVE SYS #II 800VA, M-RIN-M03B VIA E-RIN-J01A
E-NNN-D1526	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E02
E-NNN-D1525	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E01
E-NNN-D1626	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E04
E-NNN-D1625	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E03
E-QAN-B02	E-QAN-D05 CKT 2, 4, 6	LIGHTING PANEL E-QAN-D05B CTMT BLDG EL 100'

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-QAN-B03	E-QAN-D05 CKT 7, 9, 11	LIGHTING PANEL E-QAN-D05C CTMT BLDG EL 100'
E-QAN-B04	E-QAN-D05 CKT 8, 10, 12	LIGHTING PANEL E-QAN-D05D CTMT BLDG EL 140'
E-QAN-B05	E-QAN-D05 CKT 19, 21, 23	LIGHTING PANEL E-QAN-D05F CTMT BLDG EL 140'
E-QAN-B06	E-QAN-D05 CKT 13, 15, 17	LIGHTING PANEL E-QAN-D05E CTMT BLDG EL 140'
E-QBN-B01	E-QBN-D91 CKT 19, 21, 23	LIGHTING PANEL E-QBN-D73A CTMT BLDG EL 100'
E-QBN-B02	E-QBN-D91 CKT 20, 22, 24	LIGHTING PANEL E-QBN-D73B CTMT BLDG EL 140'
E-NHN-D1514	E-NHN-M1526	TO OPERATION CAMERA JB# ~~1~~ 2
E-RCN-D0101	E-NGN-L11C2	PZR BU HTR M-RCE-B07, B13, A01
E-NAN-D2614	E-NHN-M2618	TO OPERATION CAMERA JB# ~~2~~ 1
E-RCN-D0102	E-NGN-L11C2	PZR BU HTR M-RCE-B03, A09, A15
E-RCN-D0302	E-NGN-L11C3	PZR BU HTR M-RCE-B04, A11, A16
E-RCN-D0301	E-NGN-L11C3	PZR BU HTR M-RCE-A02, A07, A13
E-RCN-D0202	E-NGN-L12C2	PZR BU HTR M-RCE-B06, B12, A18
E-RCN-D0201	E-NGN-L12C2	PZR BU HTR M-RCE-B16, A04, A08
E-RCN-D0402	E-NGN-L12C3	PZR BU HTR M-RCE-B15, A03, A10
E-RCN-D0401	E-NGN-L12C3	PZR BU HTR M-RCE-A17, A06, A12

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TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NAN-S01M	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01A (C.E. NO. 1A)
E-NAN-S01L	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01C (C.E. NO. 2A)
E-NAN-S02L	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01B (C.E. NO. 1B)
E-NAN-S02-M	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01D (C.E. NO. 2B)
E-NGN-L03C2	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01C
E-NGN-L03C3	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01D
E-NGN-L03D2	FUSE IN BKR.	CTMT POLAR CRANE M-ZCN-G01
E-NGN-L06C2	E-NGN-B06C2 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01A
E-NGN-L09C4	E-NGN-B09C4 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01B
E-NGN-L10C2	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN-E01A
E-NGN-L10C3	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN- E01B
E-NGN-L10C4	FUSE IN BKR.	CONTROL PANEL CEDM M-G SET J-SFN-C02B
E-NGN-L11C4	E-NGN-L1182	PROPORTIONAL HTR BANK M-RCE-B2, B8, B14
E-NGN-L12C4	E-NGN-L12B2	PROPORTIONAL HTR BANK M-RCE-B5, B11, B17
CEA 06 CB101	F101, F102, F103	CEA 06
CEA 08 CB102	F104, F105, F106	CEA 08
CEA 10 CB103	F107, F108, F109	CEA 10

PROOF AND REVIEW

TABLE 4.8-2

ADDITIONAL RELIABILITY ACTIONS

No. of failures in last 20 valid test	No of failures in last 100 valid tests	Action
3	6	Within 14 days prepare and maintain a report for NRC audit describing the diesel generator reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	Declare the diesel generator inoperable. Perform a requalification test program for the affected diesel generator. Requalification test program requirements are indicated in Attachment 2 to this table.

ATTACHMENT 1 TO TABLE 4.8-2

REPORTING REQUIREMENT

As a minimum the Reliability Improvement Program report for NRC audit shall include:

a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

b) analysis of failures and determination of root causes of failures

c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

e) the schedule for implementation of each action from d) above

f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

Once a licensee has prepared and maintain an initial report detailing the diesel generator reliability improvement program at his site, as defined above, the licensee need prepare only a supplemental report within 14 days after each failure during a valid demand for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

In addition to the above, submit a yearly data report on the diesel generator reliability.

ATTACHMENT 2 TO TABLE 4.8-2
DIESEL GENERATOR REQUALIFICATION PROGRAM

(1) Perform seven consecutive successful demands without a failure within 30 days of diesel generator being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of diesel generator of being restored to operable status.

(2) If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator of being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of being restored to operable status.

(3) If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive successful demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

(4) Following the second failure during the requalification test program, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

(5) During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

(a) The number of failures in the last 20 valid demands is less than 5.

(b) The number of failures in the last 100 valid demands is less than 11.

(c) In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exceptior will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above).

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above the provisions of those criteria alone will prevail.

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 12 CB104	F110, F111, F112	CEA 12
CEA 07 CB101	F101, F102, F103	CEA 07
CEA 09 CB102	F104, F105, F106	CEA 09
CEA 11 CB103	F107, F108, F109	CEA 11
CEA 13 CB104	F110, F111, F112	CEA 13
CEA 74 CB101	F101, F102, F103	CEA 74
CEA 76 CB102	F104, F105, F106	CEA 76
CEA 78 CB103	F107, F108, F109	CEA 78
CEA 80 CB104	F110, F111, F112	CEA 80
CEA 75 CB101	F101, F102, F103	CEA 75
CEA 77 CB102	F104, F105, F106	CEA 77
CEA 79 CB103	F107, F108, F109	CEA 79
CEA 81 CB104	F110, F111, F112	CEA 81
CEA 22 CB101	F101, F102, F103	CEA 22
CEA 24 CB102	F104, F105, F106	CEA 24
CEA 26 CB103	F107, F108, F109	CEA 26
CEA 28 CB104	F110, F111, F112	CEA 28
CEA 23 CB101	F101, F102, F103	CEA 23
CEA 25 CB102	F104, F105, F106	CEA 25

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 27 CB103	F107, F108, F109	CEA 27
CEA 29 CB104	F110, F111, F112	CEA 29
CEA 34 CB101	F101, F102, F103	CEA 34
CEA 36 CB102	F104, F105, F106	CEA 36
CEA 38 CB103	F107, F108, F109	CEA 38
CEA 40 CB104	F110, F111, F112	CEA 40
CEA 35 CB101	F101, F102, F103	CEA 35
CEA 37 CB102	F104, F105, F106	CEA 37
CEA 39 CB103	F107, F108, F109	CEA 39
CEA 41 CB104	F110, F111, F112	CEA 41
CEA 55 CB101	F101, F102, F103	CEA 55
CEA 58 CB102	F104, F105, F106	CEA 58
CEA 61 CB103	F107, F108, F109	CEA 61
CEA 64 CB104	F110, F111, F112	CEA 64
CEA 54 CB101	F101, F102, F103	CEA 54
CEA 57 CB102	F104, F105, F106	CEA 57
CEA 60 CB103	F107, F108, F109	CEA 60
CEA 63 CB104	F110, F111, F112	CEA 63
CEA 56 CB101	F101, F102, F103	CEA 56
CEA 59 CB102	F104, F105, F106	CEA 59

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 62 CB103	F107, F108, F109	CEA 62
CEA 65 CB104	F110, F111, F112	CEA 65
CEA 66 CB101	F101, F102, F103	CEA 66
CEA 68 CB102	F104, F105, F106	CEA 68
CEA 70 CB103	F107, F108, F109	CEA 70
CEA 72 CB104	F110, F111, F112	CEA 72
CEA 67 CB101	F101, F102, F103	CEA 67
CEA 69 CB102	F104, F105, F106	CEA 69
CEA 71 CB103	F107, F108, F109	CEA 71
CEA 73 CB104	F110, F111, F112	CEA 73
CEA 02 CB101	F101, F102, F103	CEA 02
CEA 03 CB102	F104, F105, F106	CEA 03
CEA 04 CB103	F107, F108, F109	CEA 04
CEA 05 CB104	F110, F111, F112	CEA 05
CEA 42 CB101	F101, F102, F103	CEA 42
CEA 43 CB102	F104, F105, F106	CEA 43
CEA 44 CB103	F107, F108, F109	CEA 44
CEA 45 CB104	F110, F111, F112	CEA 45
CEA 82 CB101	F101, F102, F103	CEA 82
CEA 83 CB102	F104, F105, F106	CEA 83

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 84 CB103	F107, F108, F109	CEA 84
CEA 85 CB104	F110, F111, F112	CEA 85
CEA 18 CB101	F101, F102, F103	CEA 18
CEA 19 CB102	F104, F105, F106	CEA 19
CEA 20 CB103	F107, F108, F109	CEA 20
CEA 21 CB104	F110, F111, F112	CEA 21
CEA 86 CB101	F101, F102, F103	CEA 86
CEA 87 CB102	F104, F105, F106	CEA 87
CEA 88 CB103	F107, F108, F109	CEA 88
CEA 89 CB104	F110, F111, F112	CEA 89
CEA 14 CB101	F101, F102, F103	CEA 14
CEA 15 CB102	F104, F105, F106	CEA 15
CEA 16 CB103	F107, F108, F109	CEA 16
CEA 17 CB104	F110, F111, F112	CEA 17
CEA 46 CB101	F101, F102, F103	CEA 46
CEA 48 CB102	F104, F105, F106	CEA 48
CEA 50 CB103	F107, F108, F109	CEA 50
CEA 52 CB104	F110, F111, F112	CEA 52
CEA 47 CB101	F101, F102, F103	CEA 47
CEA 49 CB102	F104, F105, F106	CEA 49

PROOF AND REVIEW

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
CEA 51 CB103	F107, F108, F109	CEA 51
CEA 53 CB104	F110, F111, F112	CEA 53
CEA 30 CB101	F101, F102, F103	CEA 30
CEA 31 CB102	F104, F105, F106	CEA 31
CEA 32 CB103	F107, F108, F109	CEA 32
CEA 33 CB104	F110, F111, F112	CEA 33
CEA 01 CB101	F101, F102, F103	CEA 01

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES -

PRIMARY DEVICE NUMBER	BACK UP DEVICE NUMBER	SERVICE DESCRIPTION
E-PHA-D33-03	E-PHA-M3333	INDICATING LIGHTS FOR VLV J-SIA-UV-634
E-PHA-D33-04	E-PHA-M3333	INDICATING LIGHTS FOR VLV J-SIA-644
E-PHB-D36-01	E-PHB-M3639	INDICATING LIGHTS FOR VLV J-SIB-UV-614
E-PHB-D36-02	E-PHB-M3639	INDICATING LIGHTS FOR VLV J-SIB-UV-624
E-NHN-D28-04	E-NHN-M2831	CONTAINMENT PRE ACCESS NORMAL AFU MOTOR SPACE HEATER FOR M-MCN-F01AH.
E-NHN-D28-14	E-NHN-M2831	FLOW SWITCH J-HCN-FSL-29 FOR DUCT HEATERS M-HCN-E01A AND B
E-NHN-D28-16	E-NHN-M2831	CONTAINMENT AHU DUCT HEATERS M-HCN-E01A AND B TEMPERATURE CONTROL J-HCN-TC-29

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D28-18	E-NHN-M2831	FLOW SWITCH J-HCN-FSL-31 FOR DUCT HEATERS M-HCN-E01C AND D
E-NHN-D13-04	E-NHN-M1330	CONTAINMENT ACU DUCT HEATERS M-HCN-E01C AND D TEMPERATURE CONTROLLER J-HCN-TC-31
NHN-D13-22	E-NHN-M1330	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01AH
E-NHN-D15-01	E-NHN-M1527	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01BH
E-NHN-D15-02	E-NHN-M1527	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01DH
E-NHN-D15-06	E-NHN-M1527	CONTAINMENT PREACCESS NORMAL AFU FAN MOTOR SPACE HEATER M-HCN-F01C
E-NHN-D10-01	E-NHN-M1028	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01AH

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D10-02	E-NHN-M1028	REACTOR COOLANT PUMP MOTOR SPACE HEATER CONTACTOR M-RCE-P01CH
E-NHN-D10-20	E-NHN-M1028	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01B1
E-NHN-D19-05	E-NHN-M1915	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02AH
E-NHN-D19-06	E-NHN-M1915	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02CH
E-NHN-D19-07	E-NHN-M1915	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01AH
E-NHN-D19-08	E-NHN-M1915	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01CH
NHN-D19-10	E-NHN-M1415	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03AH

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D19-12	E-NHN-M1915	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03CH
E-NHN-D20-05	E-NHN-M2014	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02BH
E-NHN-D20-06	E-NHN-M2014	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02DH
E-NHN-D20-07	E-NHN-M2014	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01DH
E-NHN-D20-08	E-NHN-M2014	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01BH
E-NHN-D20-10	E-NHN-M2014	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03BH
E-NHN-D20-12	E-NHN-M2014	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03DH

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MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8-3 shall be bypassed continuously or under accident conditions, as applicable, by an OPERABLE device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required valves which are continuously bypassed shall be verified to be bypassed following testing during which the thermal overload protection was temporarily placed in force.

PROOF AND REVIEW

TABLE 3.8-3

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-647	HPSI A Flow Control to Reactor Coolant Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-637	HPSI A Flow Control to Reactor Coolant Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-604	HPSI Pump A Long Term Cooling Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-609	HPSI Pump B Long Term Cooling Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-657	Shutdown Clg. Temp. Control Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-658	Shutdown Clg. Temp. Control Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-685	LPSI - Ctmt Spray Pump Cross Connect A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-694	LPSI- Ctmt Spray Pump Cross Connect B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-686	Ctmt Spray A Cross Connect Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-696	Ctmt Spray B Cross Connect Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-688	Shutdown Clg. Heat Exchange A Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-693	Shutdown Clg. Heat Exchange B Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-617	HPSI A Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-627	HPSI A Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-645	LPSI Flow Control To React Coolant 1B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-635	LPSI Flow Control To React Coolant 1A Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-644	Safety Injection Tank 1B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-634	Safety Injection Tank 1A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-616	HPSI B Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-626	HPSI B Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-636	HPSI B Flow Control To React Coolant 1A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-646	HPSI B Flow Control To React Coolant 1B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-655	Shutdown Clg. Ctmt Isolation Loop 1 Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-656	Shutdown Clg. Ctmt Isolation Loop 2 Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-664	Ctmt Spray Pump A To Refueling Water Tank Isolation Vlv.	Safety Injection Shutdown Clg. Sys.

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TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIB-UV-665	Ctmt Spray Pump B To Refueling Water Tank Isolation Vlv.	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-615	LPSI Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-625	LPSI B Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-666	HPSI Pump A to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-667	HPSI Pump B to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-669	LPSI Pump A To Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-668	LPSI Pump B to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-672	Ctmt Spray Control Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-671	Ctmt Spray Control Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-674	Ctmt Sump Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-676	Ctmt Sump Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-651	Shutdown Clg. Isolation Loop 1 Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-652	Shutdown Clg. Isolation Loop 2 Valve	Safety Injection Shutdown Clg. Sys.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-673	Ctmt Sump Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-675	Ctmt Sump Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-614	Safety Injection Tank 2A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-624	Safety Injection Tank 2B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-684	Shutdown Clg. Heat Exchange Isolation Train A	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-689	Shutdown Clg. Heat Exchange Isolation Train B	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-683	LPSI Pump A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-692	LPSI Pump B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-691	Shutdown Clg. Loop 2 Warm-Up Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-690	Shutdown Clg. Loop 1 Warm-Up Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-698	HPSI Pump A Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-699	HPSI Pump B Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-306	LPSI Pump A Header Discharge Valve	Safety Injection Shutdown Clg. Sys.

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TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIB-HV-307	LPSI Pump B Header Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-687	Ctmt Spray Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-695	Ctmt Spray Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-678	Shutdown Clg. Heat Exchange Isolation Train A	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-679	Shutdown Clg. Heat Exchange Isolation Train B	Safety Injection Shutdown Clg. Sys.
J-SIC-UV-653	Shutdown Clg. Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SID-UV-654	Shutdown Clg. Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-EWA-UV-65	ECW Loop A To/From NCW Cross Tie Valve	Essential Cooling Water System
J-EWA-UV-145	ECW Loop A To/From NCW Cross Tie Valve	Essential Cooling Water System
J-CTA-HV-1	Condensate Tank to Aux. Feedwater Pump Valve	Condensate Transfer & Storage Sys.
J-CTA-HV-4	Condensate Tank to Aux. Feedwater Pump Valve	Condensate Transfer & Storage Sys.
J-SGA-UV-134	SG-1 Aux. Feedwater Pump A Steam Supply	Main Steam System
J-SGA-UV-138	SG-2 Aux. Feedwater Pump A Steam Supply	Main Steam System

PROOF AND REVIEW

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-NCB-UV-401	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-NCA-UV-402	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-NCB-UV-403	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-AFB-HV-30	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-HV-31	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-UV-34	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-UV-35	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFA-HV-32	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFA-UV-37	Aux. Feedwater Isolation Valve	Auxiliary Feed-water System
J-AFC-UV-36	Aux. Feedwater Isolation Valve	Auxiliary Feed-water System
J-AFC-HV-33	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-CPA-UV-2A	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-CPB-UV-3B	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-CPA-UV-2B	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-CPB-UV-3A	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-CPA-UV-4A	Ctmt Purge Power Access Mode Isolation Valve	Containment Purge System
J-CPB-UV-5B	Ctmt Purge Power Access Mode Isolation Valve	Containment Purge System
J-CPA-UV-4B	Ctmt Purge Power Access Mode Isolation Valve	Containment Purge System
J-CPB-UV-5A	Ctmt Purge Power Access Mode Isolation Valve	Containment Purge System
J-WCA-UV-62	Normal Chill Water Return Ctmt Isolation	Chilled Water System
J-WCB-UV-63	Normal Chill Water Supply Ctmt Isolation	Chilled Water System
J-WCB-UV-61	Normal Chill Water Return Ctmt Isolation	Chilled Water System
J-RDA-UV-23	Ctmt Radwaste Sumps Internal Isolation	Radioactive Waste Drain System
J-HPA-UV-3	H_2 Ctmt Train A Downstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPA-UV-5	H_2 Ctmt Train A Return Isolation Valve	Containment Hydrogen Control Sys.
J-HPB-UV-4	H_2 Ctmt Train B Downstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPB-UV-6	H_2 Ctmt Train B Return Isolation Valve	Containment Hydrogen Control Sys.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-HPB-UV-2	H_2 Ctmt Train B Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPA-UV-1	H_2 Ctmt Train A Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-GRA-UV-1	Radioactive Drain Tk Gas Surge Hdr Internal Containment Isolation	Gaseous Radwaste System

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2150 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification ~~4.6.3.2.~~

4.9.9.

PROOF AND REVIEW

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

PROOF AND REVIEW

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REFUELING OPERATIONS

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3590 (3990)* pounds and an overload cut off limit of less than or equal to 1556 ~~(1736)~~* pounds for the fuel mast REFUELING MACHINE 1727.

b. A minimum capacity of 2000 pounds and an overload cut off limit of less than or equal to 1651 (1831)* pounds for the CEA mast.

APPLICABILITY: During movement of CEAs or fuel assemblies within the ~~reactor pressure vessel~~ REFUELING CAVITY.

ACTION:

a. With the above requirements for the fuel mast REFUELING MACHINE not satisfied, suspend use of the fuel mast REFUELING MACHINE from operations involving the movement of fuel assemblies.

b. With the above requirements for the CEA mast not satisfied, suspend use of the CEA mast from operations involving the movement of CEAs.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The fuel mast REFUELING MACHINE used for movement of fuel assemblies shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3590 (3990)* pounds and demonstrating an automatic load cut off when the fuel mast REFUELING MACHINE load exceeds 1556 ~~(1736)~~* 1727 pounds.

4.9.6.2 The CEA mast used for movement of CEAs shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 2000 pounds and demonstrating an automatic load cut off when the CEA mast load exceeds 1651 (1831)* pounds.

*For initial fuel load only.

PROOF AND REVIEW

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

PROOF AND REVIEW

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge valve isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

PROOF AND REVIEW

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

PROOF AND REVIEW

CEAs

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the fuel seated in the reactor pressure vessel.

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of CEAs.

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REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least ~~22~~ 22'-8" feet of water shall be maintained over the top of ~~irradiated fuel assemblies seated in~~ the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel building essential ventilation systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel building essential ventilation system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel building essential ventilation system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel building essential ventilation system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel building essential ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building essential ventilation system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel building essential ventilation systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

PROOF AND REVIEW

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52; Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, heaters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the fuel building at a slight negative pressure relative to the outside atmosphere during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.0% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

PROOF AND REVIEW

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2 and 3.*

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

PROOF AND REVIEW

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

PROOF AND REVIEW

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During STARTUP and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

PROOF AND REVIEW

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

3/4.10.5 MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature for criticality limits of Specification 3.1.1.4 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 300°F provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Variable Overpower trip channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation required by Specification 3.4.8 except that the core critical line shown on Figure 3.4-2 does not apply.

APPLICABILITY: MODE 2*.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

4.10.5.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 300°F at least once per hour.

4.10.5.4 Each Logarithmic Power Level and Variable Overpower channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

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3/4.10.6 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.10.6 The safety injection tank isolation valve requirement of Specification 3.5.1a. may be suspended during partial stroke testing of the low pressure safety injection check valves (SI-114, SI-124, SI-134, SI-144) provided:

a. That power to the isolation valve is restored and the SIAS signal is not overridden.

b. Only one isolation valve at a time is closed during the testing for no longer than 1 hour.

c. That the valve is KEY LOCKED opened with power removed before the next isolation valve is closed.

APPLICABILITY:

While partial stroke testing of the low pressure injection check valves during normal plant operation.

ACTION:

If the requirement of Specification 3.5.1a. was suspended to perform the Specification 3.10.6 partial stroke test and if any of the Specification 3.10.6 requirements are not met during the Specification 3.10.6 partial stroke testing, the Limiting Condition for Operation shall revert to Specification 3.5.1 and the 3.5.1 ACTION shall be applicable.

SURVEILLANCE REQUIREMENTS

4.10.6.1 A valve alignment shall be performed within 4 hours following completion of testing to verify that all valves operated during this testing are restored to their normal positions and that power is removed to the SIT isolation valves.

3/4.10.7 SPENT FUEL POOL LEVEL

LIMITING CONDITION FOR OPERATION

3.10.7 The borated water source of Specifications 3.1.2.5a. and 3.1.2.6a. may be suspended during initial fuel load and startup provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: MODES 2, 3, 4, 5, and 6.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.7.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.7.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

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3/4.10.8 SAFETY INJECTION TANK PRESSURE

LIMITING CONDITION FOR OPERATION

3.10.8 The safety injection tank (SIT) pressure of Specification 3.5.1d. may be suspended for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b. and pressurized to 175 to 225 psig below the RCS pressure; OR NOT TO GO BELOW 254 PSIG

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2 and 3.

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.8, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.8.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.8.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specification 3.5.1b.

b. All the SITs pressures meet the requirements of Specification 3.10.8.

c. The valve alignment from the SITs to the RCS has not changed.

PROOF AND REVIEW

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGES TO ONSITE EVAPORATION PONDS CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the lower limit of detectability (LLD) defined as 5×10^{-7} μCi/ml for the principal gamma emitters or 1×10^{-6} μCi/ml for I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

When any secondary system liquid waste discharge pathway concentration determined in accordance with the surveillance requirements given below exceeds the specified LLD, divert that discharge pathway to the liquid radwaste system without delay.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes collected in the chemical waste neutralizer tank shall be sampled and analyzed prior to their batchwise discharge to the onsite evaporation pond in accordance with the sampling and analysis program specified in Table 4.11-1.

4.11.1.1.2 With the concentration of radioactive material in the chemical waste neutralizer tank exceeding the specified LLD, sample and analyze other secondary system discharge pathways in accordance with the sampling and analysis program specified in Table 4.11-1.

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TABLE 4.11-1

SECONDARY SYSTEM LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

SECONDARY SYSTEM LIQUID RELEASE PATHWAY	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/mL)
A. Batch discharges[b]				
1. Chemical Waste Neutralizer Tank	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
2. Steam Generator Blowdown Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
3. Condensate Polishing Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
B. Continuous Releases[d]				
1. Turbine Building Sump*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
2. Condenser Area Sumps*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$

*Sampling and analysis for pathways 2 and 3 under batch discharges and 1 and 2 under continuous releases are required only when concentration for chemical waste neutralizer tank pathway exceeds the LLD.

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\mathrm{LLD} = \frac{4.66\, s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

[b] PRIOR TO DISCHARGE EACH BATCH SHALL BE ISOLATED AND SAMPLED IN A REPRESENTATIVE MANNER

TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

DELETE

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

PROOF AND REVIEW

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. 500

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank or the reactor makeup water tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

PROOF AND REVIEW

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figures ~~5.1-1~~ and 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For I-131 AND I-133 ~~all radioiodines,~~ for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to ~~radioactive materials, other than noble gases~~, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

I-131, I-133, TRITIUM AND ALL RADIONUCLIDES IN PARTICULATE FORM WITH HALF-LIVES GREATER THAN 8 DAYS

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[G]	1×10^{-4}
B. Containment Purge	P Each Purge[b,c] Grab Sample	P Each Purge[b,c]	Principal Gamma Emitters[G]	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Vacuum Pump Exhaust 2. Plant Vent 3. Fuel Bldg. Exhaust	M[b,E] Grab Sample	M[b]	Principal Gamma Emitters[G]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[F]	W[D] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[F]	W[D] Particulate Sample	Principal Gamma Emitters[G] (I-131, Others)	1×10^{-11}
	Continuous[F]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[F]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
D. All Radwaste Types as listed in A., B., and C. above.	Continuous[F]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

PROOF AND REVIEW

PROOF AND REVIEW

TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detectable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between the midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report (ARH-2537 (June 22, 1972).

PROOF AND REVIEW

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TABLE 4.11-2 (Continued)

TABLE NOTATION

b~~Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.~~

cSampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

~~dTritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.~~

D ~~e~~ Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). ~~Sampling shall also be performed at least once per 24 hours for at least 7 days following each SHUTDOWN, STARTUP, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing.~~ [~~When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.~~] STET

E ~~f~~ Tritium grab samples shall be taken at least MONTHLY ~~once per 7 days~~ from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

F ~~g~~ The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

G ~~h~~ The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

b ANALYSES SHALL ALSO BE PERFORMED FOLLOWING SHUTDOWN, STARTUP, OR A THERMAL POWER CHANGE EXCEEDING 15% OF THE RATED THERMAL POWER WITHIN A 1 HOUR PERIOD UNLESS 1) ANALYSIS SHOWS THAT THE DOSE EQUIVALENT I-131 CONCENTRATION IN THE PRIMARY COOLANT HAS NOT INCREASED MORE THAN A FACTOR OF 3; AND (2) THE NOBLE GAS ACTIVITY MONITOR ON THE PLANT VENT SHOWS THAT EFFLUENT ACTIVITY HAS NOT INCREASED BY MORE THAN A FACTOR OF 3. IF THE ASSOCIATED NOBLE GAS VENT MONITOR IS INOPERABLE, samples must be obtained as soon as possible ANALYSES SHALL BE PERFORMED WITHIN A FOUR HOUR PERIOD. THIS REQUIREMENT DOES NOT APPLY TO THE FUEL BUILDING EXHAUST.

PROOF AND REVIEW

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figures 5.1-1 and 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site (see Figures 5.1-1 and 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, from the site (see Figures 5.1-1 and 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

PROOF AND REVIEW

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within ~~1~~ 6 hour and less than or equal to 2% by volume within 48 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.9.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 170,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added to the tank and the quantity of radioactivity contained in the tank is less than or equal to one-half of the above limit; otherwise, determine the quantity of radioactive material contained in the tank at least once per 24 hours. DURING ADDITION.

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3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

PROOF AND REVIEW

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in gaseous effluents exceeding twice the limits of Specifications 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4a.

PROOF AND REVIEW

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

PROOF AND REVIEW

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1, and the detection capabilities required by Table 4.12-1.

PROOF AND REVIEW

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS	NUMBER AND APPROXIMATE LOCATION OF SAMPLES[a]
Airborne			
Radioiodine and particulates	Continuous sampling collected weekly[c]	Gross beta weekly; I-131 weekly; gamma spectrum monthly; composite of filters[d,e]	Samples from 5 locations: 3 samples at or near the SITE BOUNDARIES, in different sectors of the highest calculated annual average ground level D/Q.*
			1 sample from areas of special interest, which is from the vicinity of a community having the highest calculated annual average D/Q.
			1 sample from a control location 15-30 km (10-20 mi) distant and in the least prevalent wind direction.[f]
Direct radiation[b]	Quarterly	Gamma dose quarterly	40 stations with two or more dosimeters for measuring dose rate continuously, placed as follows: an inner ring of stations at the site boundary and an outer ring in the 4-to-5 mi range from the site with a station in each sector of each ring, except the WNW sector, which is inaccessible (16 sectors x 2 rings minus 1 = 31 stations). 7 additional stations are in local schools and population centers; 2 other stations are used as controls.

*D/[illegible]ers to average annual relative ground deposition rate.

PROOF AND REVIEW

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS	NUMBER AND APPROXIMATE LOCATION OF SAMPLES[a]
Waterborne			
Surface	Monthly composite of weekly grab sample	Gamma spectrum monthly; tritium quarterly	Water storage reservoir evaporation pond
Ground	Quarterly grab sample	Tritium and gamma spectrums quarterly	2 onsite wells[g]
Drinking (well)	Monthly composite of weekly grab sample	Gross beta and gamma spectrums monthly; tritium quarterly[h]	3 wells from surrounding residences
Ingestion			
	Semimonthly for animals on pasture; otherwise, monthly	Gamma spectrum and radioiodine semi-monthly or monthly[h]	Local dairy
Food products	At harvest[i]	Gamma spectrum and radioiodine monthly[h]	Local farms

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

[a]The number, media, frequency, and location of sampling may vary from site to site. It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and submitted for acceptance. Actual locations (distance and direction) from the site shall be provided in a table and a figure in the ODCM. Refer to Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants."

[b]Regulatory Guide 4.13 provides guidance for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter may be considered to be one phosphor, and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

[c]Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

[d]Particulate sample filters shall be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than 10 times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.

[e]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[f]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

[g]Groundwater samples should be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

[h]The dose shall be calculated for the maximum organ and age group, using the methodology contained in Regulatory Guide 1.109, Rev. 1, and the actual parameters particular to the site.

[i]If harvest occurs more than once a year, sampling should be performed during each discrete harvest. If harvest occurs continuously, sampling should be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*			
Mn-54	1,000			
Fe-59	400			
Co-58	1,000			
Co-60	300			
Zn-65	300			
Zr-Nb-95	400			
I-131	2	0.9	3	100
Cs-134	30	10	60	1,000
Cs-137	50	20	70	2,000
Ba-La-140	200		300	

*For drinking water samples. This is 40 CFR Part 141 value.

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TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m^3)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg,wet)
Gross beta	4	0.01		
H-3	2000			
Mn-54	15			
Fe-59	30			
Co-58,-60	15			
Zn-65	30			
Zr-95	30			
Nb-95	15			
I-131	1[c]	0.07	1	60
Cs-134	15	0.05	15	60
Cs-137	18	0.06	18	80
Ba-140	60		60	
La-140	15		15	

Note: This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

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TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]Guidance for detection capabilities for thermoluminescent dosimeters used for environmental measurements is given in Regulatory Guide 4.13.

[b]Table 4.12-1 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of this guide, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocuries per unit mass or volume).

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration)

V is the sample size (in units of mass or volume)

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt for environmental samples is the elapsed time between sample collection (or end of the sample collection period) and time of counting

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TABLE 4.12-1 (Continued)

TABLE NOTATION

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[c] LLD for drinking water samples.

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3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1 shall be followed, including analysis of control samples.

PROOF AND REVIEW

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

PROOF AND REVIEW

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

PROOF AND REVIEW

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two containment spray systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required containment spray systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

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4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual surveillance requirements. Surveillance requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems, or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems, or components OPERABLE, when such items are found or known to be inoperable although still meeting the surveillance requirements.

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4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

PROOF AND REVIEW

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis. With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and A 4% $\Delta k/k$ SHUTDOWN MARGIN requirement~~s~~ ~~are~~ is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

P

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) an emergency power supply from OPERABLE diesel generators. THE 26 GPM VALUE IS BASED ON NOMINAL FLOW OF ONE CHARGING PUMP LESS THE REACTOR COOLANT PUMP BLEEDOFF FLOW

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

BORATION SYSTEMS (Continued)

The values of water volumes, temperatures, and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analysis remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in MODES 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.2.7 BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at any time, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

With only one startup channel high neutron flux alarm OPERABLE while in MODE 3, 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. ~~This provides a diverse and redundant method of detection of boron dilution, with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality~~.

THIS PROVIDES ALTERNATE METHODS OF DETECTION OF BORON DILUTION, WITH SUFFICIENT TIME FOR TERMINATION OF THE EVENT BEFORE COMPLETE LOSS OF SHUTDOWN MARGIN AND RETURN TO CRITICALITY

b. Both startup channel high neutron flux alarms not operating:

When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, simultaneous use of boronometer and RCS sampling to monitor the RCS boron concentration provides diverse and redundant indications of an inadvertent boron dilution. ~~This will allow detection with sufficient time for termination of boron dilution before complete loss of SHUTDOWN MARGIN and return to criticality~~.

THEREFORE, EITHER SIMULTANEOUS USE OF THE BORONOMETER AND RCS SAMPLING OR INDEPENDENT COLLECTION AND ANALYSIS OF TWO RCS SAMPLES TO MONITOR

THE RCS BORON CONCENTRATION PROVIDES ALTERNATE INDICATIONS OF AN INADVERTENT BORON DILUTION.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs, and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO

MOVABLE CONTROL ASSEMBLIES (Continued)

and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{cold} greater than or equal to 552°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches or greater.

The establishment of LSSS and LCOs requires that the expected long- and short-term behavior of the radial peaking factors be determined. The long-term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short-term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

PLUS THOSE ASSOCIATED WITH THE CPC STARTUP TEST ACCEPTANCE CRITERIA

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{tilt}/P_{untilt})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_0)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_0 is the azimuthal core location of maximum tilt

PROOF AND REVIEW

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the

3/4.3.2 AUXILIARY SPRAY VALVES

Cycling of the auxiliary spray valves during normal plant cooldown will insure the operability of the valves while not causing any additional impact on pressurizer spray nozzle usage factor.

PROOF AND REVIEW

individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; ~~and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.~~

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

PROOF AND REVIEW

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

ACTUATED

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are estimated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

PROOF AND REVIEW

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

DELETE

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment, or structures.

PROOF AND REVIEW

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.231 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 4000 gpm will circulate one equivalent Reactor Coolant System volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to ~~241°F~~ 255 °F during cooldown or ~~281~~ 295°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., there is no direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 STEAM GENERATORS

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 gpm per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Containment sump flow is provided by monitoring the rate of sump level increase prior to the sump being pumped down, and is alarmed at the equivalent of 1 gpm leakage into the sump. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

PROOF AND REVIEW

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value. A threshold value of less than 1 gpm is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for both steam generators ensures that the dosage contribution from the tube leakage will be limited to less than Part 100 guidelines for infrequent and limiting fault events. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude may be indicative of an impending failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss-of-offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Palo Verde site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. ~~Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location~~.

BUT BECAUSE OF HIGHER NEUTRON FLUENCE MAY BE CONTROLLING.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp~~;~~. ~~therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined.~~ Consequently, ~~for the cases in which the outer wall of the vessel becomes the stress controlling location,~~ each heatup rate of interest must be analyzed on an individual basis FOR BOTH THE INNER AND OUTER WALLS.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and ~~copper content of the material in question~~ residual element content, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a)(b) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_V ENERGY FOR LONGITUDINAL DIRECTION-ft lb
128-101	M-6703-1	SA 508-CL2	Inlet Nozzle	-20	0	+20	+60	N.A.
128-101	M-6703-2	SA 508-CL2	Inlet Nozzle	+10	+10	-25	+10	N.A.
128-101	M-6703-3	SA 508-CL2	Inlet Nozzle	-10	-10	-27	+18	N.A.
128-101	M-6703-4	SA 508-CL2	Inlet Nozzle	0	0	+5	+42	N.A.
131-102	M-4307-1	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
131-102	M-4307-2	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
128-501	M-6708-1	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-2	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-3	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-501	M-6708-4	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-301	M-4304-1	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
128-301	M-4304-2	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
131-101	M-6712-1	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-2	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-3	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
131-101	M-6712-4	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
126-101	M-6705-1	SA 508-CL2	Vessel Flange	-70	-70	-78	-28	N.A.
106-101	M-6706-1	SA 508-CL2	Closure Head Flange	-70	-70	-80	-54	N.A.

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values.
** = Average of three test results.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB., Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

(PLATES)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_v ENERGY FOR LONGITUDINAL DIRECTION-ft lb
142-102	M-4311-1	SA 533-GRB-CL1	Lower Shell Plate	-10	-10	-6	+40	134
142-102	M-4311-2	SA 533-GRB-CL1	Lower Shell Plate	-40	-40	-24	-8	127
142-102	M-4311-3	SA 533-GRB-CL1	Lower Shell Plate	-20	-20	-7	+14	142
124-102	M-6701-1	SA 533-GRB-CL1	Intermed. Shell Plate	-40	+30	+44	+90	83
124-102	M-6701-2	SA 533-GRB-CL1	Intermed. Shell Plate	-50	+40	+56	+98	96
124-102	M-6701-3	SA 533-GRB-CL1	Intermed. Shell Plate	-30	+40	+39	+89	100
122-102	M-6701-4	SA 533-GRB-CL1	Upper Shell Plate	-30	+60	+82	+120	N.A.
122-102	M-6701-5	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+49	+98	N.A.
122-102	M-6701-6	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+42	+96	N.A.
102-102A	M-6709-1	SA 533-GRB-CL1	Closure Head Dome	-20	+10	+36	+66	N.A.
102-102B	M-6709-2	SA 533-GRB-CL1	Closure Head Dome	-70	-20	+4	+37	N.A.
150-102	M-6715-1	SA 533-GRB-CL1	Bottom Head Dome	-30	-30	+2	+30	N.A.
150-102	M-6715-2	SA 533-GRB-CL1	Bottom Head Dome	-40	-10	+26	+50	N.A.

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.

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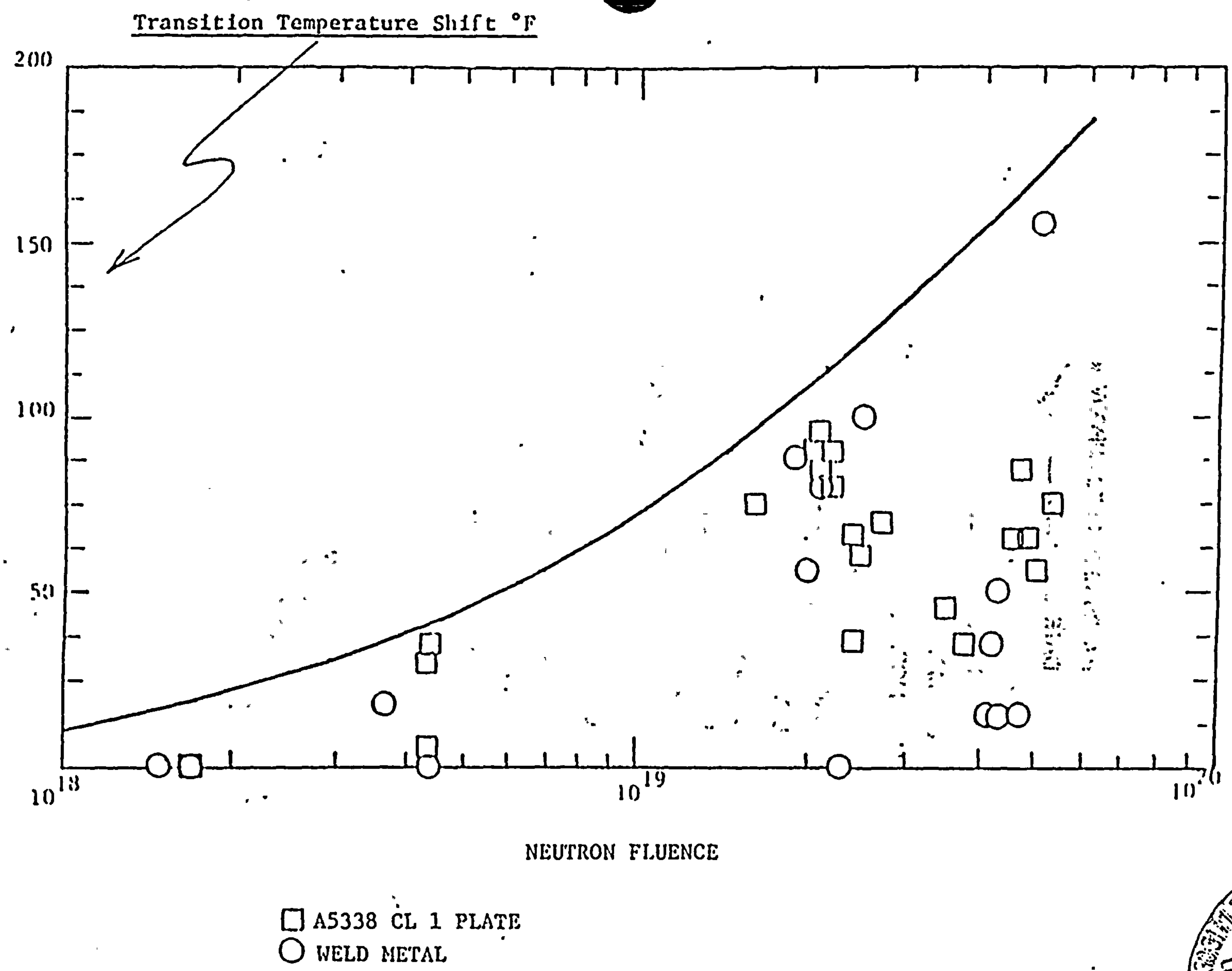

FIGURE B 3/4.4-1

NIL-DUCTILITY TRANSITION TEMPERATURE INCREASE AS A FUNCTION OF FAST (E > 1 MeV) NEUTRON FLUENCE (550°F IRRADIATION)

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 255 ~~241~~°F during cooldown and 295 ~~281~~°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

PROOF AND REVIEW

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of ~~4000~~ 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

.1 of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1750 psia the Technical Specifications require a minimum of 57% wide range corresponding

SAFETY INJECTION TANKS (Continued)

to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the RCS temperatures greater than or equal to 350°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide

PROOF AND REVIEW

ECCS SUBSYSTEMS (Continued)

assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

AT ATMOSPHERIC PRESSURE

1. The pressurizer pressure is ~~15 psia~~ 6.4
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) ~~3.2~~ gpm (to/from) the ~~2450~~ gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline. 4900

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OPERATIONS
LICENSING

REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

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3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

2.5

The maximum peak pressure expected to be obtained from a LOCA event is 49.2 psig. The limit of ~~4~~ psig for initial positive containment pressure will limit the total pressure to 49.2 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 49.2 psig in the event of a LOCA. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", June 1974.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure ADEQUATE CHEMICAL AVAILABLE TO REMOVE IODINE FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA. ~~a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes~~

PROOF AND REVIEW

~~IODINE REMOVAL SYSTEM (Continued)~~

~~the evolution of iodine and minimizes the effect of chloride stress corrosion and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety~~ analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

PROOF AND REVIEW

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1381 psig) of its design pressure of 1256 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 19.53×10^6 lb/hr which is 113% of the total secondary steam flow of 17.18×10^6 lb/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two-loop, or four-pump operation

$$SP = \left(\frac{10-N}{10}\right) \times 113$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = total number of secondary safety valves for one steam generator.

N = the number of inoperable secondary safety valves on the steam generator with the greater number of inoperable valves.

113 = the ratio of the total relieving capacity of all twenty (20) secondary safety valves (19.53×10^6 lb/hr at 1355 psig, maximum set pressure plus 3%, accumulation) over the secondary steam flow at 100% Rated Thermal Load (17,180,000 lb/hr).

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PROOF AND R

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M1341 (FUSE)	E-NHN-M1341 (FUSE)	REACTOR CAVITY FAN C DISCH DAMPER M-HCN-M02C
E-NHN-M1306 (FUSE)	E-NHN-M1306 (FUSE)	SG 2 HOT LEG BLDWN ISOL VLV J-SGE-HV-42
E-NHN-M1307 (FUSE)	E-NHN-M1307 (FUSE)	SG2 COLD LEG BLDWN ISOL VLV J-SGE-HV-44
E-NHN-M2803 (FUSE)	E-NHN-M2803 (FUSE)	CEDM ACU C INTAKE DAMPER M-HCN-M03C
E-NHN-M2804 (FUSE)	E-NHN-M2804 (FUSE)	CEDM ACU D INTAKE DAMPER M-HCN-M03D
E-NHN-M2805 (FUSE)	E-NHN-M2805 (FUSE)	SG1 COLD LEG BLDWN ISO VLV J-SGE-HV-41
E-NHN-M2806 (FUSE)	E-NHN-M2806 (FUSE)	SG 1 HOT LEG BLDWN ISO VLV J-SGE-HV-43
E-NHN-M2808 (FUSE)	E-NHN-M2808 (FUSE)	RCP 2B CONTROL BLEEDOFF VALVE J-RCE-HV-433
E-NHN-M2813 (FUSE)	E-NHN-M2813 (FUSE)	RCP 2B HI PRESS COOLER INLET VLV J-RCN-HV-449
E-NHN-M1004 (FUSE)	E-NHN-M1004 (FUSE)	RCP 1B HP COOLER INLET VLV J-RCN-HV-447
E-NHN-M1006 (FUSE)	E-NHN-M1006 (FUSE)	SG WET LAYUP RECIRC PUMP M-SGN-P01B
E-NHN-M1005 (FUSE)	E-NHN-M1005 (FUSE)	RCP 1B HP COOLER INLET VLV J-RCN-HV-451
E-NHN-M1009 (FUSE)	E-NHN-M1009 (FUSE)	RCP 2B HI PRESS COOLER INLET VLV J-RCN-HV-453
E-NHN-M1010 (FUSE)	E-NHN-M1010 (FUSE)	REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-M02B
E-PHA-M3604 (FUSE)	E-PHA-M3604 (FUSE)	SHUT DN CLG ISO LOOP 2 VLV J-SIB-UV-652
E-PHA-M3613 (FUSE)	E-PHA-M3613 (FUSE)	CTMT SUMP ISOL TRAIN-B VLV J-SIB-UV-675
E-PHA-M3619 (FUSE)	E-PHA-M3619 (FUSE)	SAFETY INJECTION TANK 1 ISOL VLV J-SIB-UV-614
E-NHN-M1904 (FUSE)	E-NHN-M1904 (FUSE)	REACTOR CAVITY NORM CLG FAN M-HCN-A03C

NOTE: THE ABOVE BACKUP PROTECTION FUSES WERE ADDED BY DCP NO. 1SE-PH-036, 2CE-PH-036, 3CE-PH-036

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TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

[illegible]MARY DEVICE [illegible]UMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M1907 (FUSE)	E-NHN-M1907 (FUSE)	CEDM NORM ACU A HEXCH OUTLET VLV J-NCN-HV-485
E-NHN-M1911 (FUSE)	E-NHN-M1911 (FUSE)	CTMT NORM ACU-C CHILLED WATER INLET VLV J-WCN-HV-5[illegible]
E-NHN-M1912 (FUSES)	E-NHNM1912 (FUSE)	CTMT NORM ACU-A CHILLED WATER INLET VLV J-WCN-HV-57
E-NHN-M2003 (FUSE)	E-NHN-M2003 (FUSE)	CTMT NORM ACU-B CHILLED WATER INLET VLV J-WCN-HV-58
E-NHN-M2004 (FUSE)	E-NHN-M2004 (FUSE)	CTMT NORM ACU-D CHILLED WATER INLET VLV J-WCN-HV-60
E-NHN-M2006 (FUSE)	E-NHN-M2006 (FUSE)	REACTOR CAVITY NORM CLG FAN M-HCN-A03B
E-NHN-M2008 (FUSE)	E-NHN-M2008 (FUSE)	CEDM NORM ACU B HEXCH OUTLET VLV J-HCN-HV-486
E-NHN-M1503 (F[illegible])	E-NHN-M1503 (FUSE)	RCP 1A CONTROL BLEEDOFF VLV J-RCE-HV-430
E-[illegible]HN-M1504 (FUSE)	E-NHN-M1504 (FUSE)	RCP 2A CONTROL BLEEDOFF VLV J-RCE-HV-432
E-NHN-M1505 (FUSE)	E-NHN-M1505 (FUSE)	RCP 1A HI PRESS COOLER INLET VLV J-RCN-HV-446
E-NHN-M1506 (FUSE)	E-NHN-M1506 (FUSE)	RCP 2A HI PRESS COOLER INLET VLV J-RCN-HV-448
E-NHN-M1507 (FUSE)	E-NHN-M1507 (FUSE)	RCP 1A HI PRESS COOLER OUTLET VLV J-RCN-HV-450
E-NHN-M1508 (FUSE)	E-NHN-M1508 (FUSE)	RCP 2A HI PRESS COOLER OUTLET VLV J-RCN-HV-452
E-NHN-M1509 (FUSE)	E-NHN-M1509 (FUSE)	REACTOR CAVITY FAN A DISCH DAMPER M-HCN-M02A
E-NHN-M1533 (FUSE)	E-NHN-M1533 (FUSE)	REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02D
E-PHB-M3811 (FUSE)	E-PHB-M3811 (FUSE)	NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61
[illegible]HB-M3816 [illegible])	E-PHB-M3816 (FUSE)	H2 CTMT TR B UPSTM SUP ISO VLV J-HPB-UV-2
E-NHN-M7102 (FUSE)	E-NHN-M7102 (FUSE)	CTMT NORM ACU A DISCH DAMPER M-HCN-M01A

NOTE: THE ABOVE BACKUP PROTECTION FUSES WERE ADDED BY DCP NO. 1SE-PH-036, 2CE-PH-036, 3CE-PH-036

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TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-NHN-M7103 (FUSE)	E-NHN-M7103 (FUSE)	CTMT NORM ACU C DISCH DAMPER M-HCN-M01C
M-HCN-E01A (FUSE)	M-HCN-E01A (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01A
M-HCN-E01B (FUSE)	M-HCN-E01B (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01B
M-HCN-E01C (FUSE)	M-HCN-E01C (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01C
M-HCN-E01D (FUSE)	M-HCN-E01D (FUSE)	CTMT NORM ACU DUCT HEATER M-HCN-E01D

NOTE: THE ABOVE BACKUP PROTECTION FUSES WERE ADDED BY DCP NO. 1SE-PH-036, 2CE-PH-036, 3CE-PH-036

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZAA-C03 (FUSE)	E-PKA-D2109	REACTOR DRAIN TANK OUTLET FSOL VLV J-CHA-UV-560
E-ZAA-C03 (FUSE)	E-PKA-D2109	SI TK RWT RTN HDR CTMT ISO VLV J-SIA-UV-682
E-ZAA-C03 (FUSE)	E-PKA-D2109	REGENERATIVE HEAT EXCH TO AUX SPRAY VLV J-CHA-HV-205
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VLV J-SSA-UV-203
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VLV J-SSA-UV-204
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISO VLV J-SSA-UV-205
E-ZAA-C04 (FUSE)	E-PKA-D2102	PRESSURIZER VENT VALVE J-RCA-HV-103
E-ZAA-C05 (FUSE)	E-PKA-D2114	STEAM GEN BLOWDOWN CTMT ISO VLV J-SGA-UV-500P
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO VLV J-SGA-UV-204
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO VLV. J-SGA-UV-211
E-ZAA-C05 (FUSES)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISO VLV J-SGA-UV-220

NOTE: THE ABOVE BACKUP PROTECTION CONTROL FUSES WERE ADDED BY DCP. NO. 1SE-PK-029, 2CE-PK-029, 3CE-PK-029

PROOF AND REVIEW

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	SECONDARY DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TK NITROGEN SPLY VLV J-SIA-HV-619
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TK NITROGEN SPLY VLV J-SIA-HV-629
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-605
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-606
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VLV J-SIA-HV-607
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VLV J-SIA-HV-608
E-ZAA-C06 (FUSE)	E-PKA-D2121	RC SYSTEM VENT TO CTMT VLV J-RCA-HV-106
E-ZAB-C03 (FUSE)	E-PKB-D2209	REGEN HEAT EXCH TO AUX SPRAY VALVE J-CHB-HV-203
E-ZAB-C03 (FUSE)	E-PKB-D2209	REACTOR COOLANT VENT VLV J-RCB-HV-102
E-ZAB-C03 (FUSE)	E-PKB-D2209	SAFETY INJ TK FILL + DRAIN VLV J-SIB-UV-611
E-ZAB-C03 (FUSE)	E-PKB-D2209	SI TK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-618
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADN MONITORING ISO VLV J-HCB-UV-44
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADN MONITORING ISO VLV J-HCB-UV-47
E-ZAB-C04 (FUSE)	E-PKB-D2202	REACTOR COOLANT VENT VLV J-RCB-HV-108
E-ZAB-C04 (FUSE)	E-PKB-D2202	SAFETY INJ TK FILL + DRAIN VLV J-SIB-UV-621
E-ZAB-C04 (FUSE)	E-PKB-D2202	SI TK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-628
E-ZAB-C05 (FUSE)	E-PKB-D2214	REACTOR COOLANT VENT VLV J-RCB-HV-109
E-ZAB-C05 (FUSE)	E-PKB-D2214	STEAM GEN BLOWDOWN CTMT ISO VLV J-SGB-UV-500R

NOTE: THE ABOVE BACKUP PROTECTION CONTROL CIRCUIT FUSES WERE ADDED BY DCP NO. 1SE-PK-027, 2CE-PK-027, 3CE-PK-027

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PROOF AND REVIEW

TABLE 3.8-2 (CONTINUED)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO VLV J-SGB-UV-222
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO VLV J-SGB-UV-224
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISO VLV J-SGB-UV-226
E-ZAB-C06 (FUSE)	E-PKB-D2221	REACTOR COOLANT VENT VLV J-RCB-HV-105
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK NITROGEN SPLY VLV J-SIB-UV-612
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK NITROGEN SPLY VLV J-SIB-UV-622
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-613
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-623
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-633
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TK VENT VLV J-SIB-HV-643
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TK NITROGEN SPLY VLV J-SIA-HV-639
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TK NITROGEN SPLY VLV J-SIA-HV-649
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO RDT VALVE J-CHA-HV-507
E-ZJA-C03 (FUSE)	E-PKA-D2111	LETDOWN LINE TO REGEN HEAT EXCH CTMT ISO VALVE J-CHA-UV-516
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO VCT VALVE J-CHA-UV-506
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TK FILL & DRAIN VLV J-SIB-UV-641
E-ZJB-C01 (FUSE)	E-PKB-D2201	SI TK CHECK VLV LEAKAGE LINE ISO VALVE J-SIB-UV-648
E-ZJB-C01 (FUSE)	E-PKB-D2201	HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-322

NOTE: THE ABOVE CONTROL CKT FUSE WERE ADDED BY DCP NO. 1SL, 2CE, 3CL-PK-027

PROOF AND REVIEW

TABLE 3.8-2 (CONTINUED)
CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES - CONTROL CIRCUITS

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	CONTROL CIRCUIT FOR
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TK NITROGEN SPLY VLV J-SIB-HV-632
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ KT NITROGEN SPLY VLV J-SIB-HV-642
E-ZJB-C03 (FUSE)	E-PKB-D2211	LETDOWN LINE TO REGEN HEAT EXCH VALVE J-CHB-UV-515
E-ZJB-C03 (FUSE)	E-PKB-D2211	SAFETY INJ TK FILL & DRAIN VLV J-SIB-UV-631
E-ZJB-C03 (FUSE)	E-PKB-D2211	SI TK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-638
E-ZJB-C03 (FUSE)	E-PKB-D2211	HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-332
E-ZAN-C01 (FUSE)	E-NKN-D4226	SEAL INJECT VALVES TO RCP J-CHE-FV-241
E-ZAN-C01 (FUSE)	E-NKN-D4224	SEAL INJECT VALVES TO RCP J-CHE-FV-242
E-ZAN-C01 (FUSE)	E-NKN-D4222	SEAL INJECT VALVES TO RCP J-CHE-FV-244
E-ZAN-C01 (FUSE)	E-NKN-D4224	POST ACDT SMPLG SYS ISO VLV J-CHN-HV-923
E-ZAN-C01 (FUSE)	E-NKN-D4224	REACTOR VESSEL SEAL DRAIN TO RDT VLV J-RCE-HV-403
E-ZAN-C01 (FUSE)	E-NKN-D4224	SI DRAIN TO REACTOR DRAIN TK VLV J-SIE-HV-661
E-ZAN-C02 (FUSE)	E-NKN-D4216	SEAL INJECT VALVES TO RCP J-CHE-FV-243
E-ZAN-C02 (FUSE)	E-NKN-D4216	REGEN HEAT EXCH TO CHARGING LINE VLV J-CHE-PDV-240
E-PGB-L32E2 (FUSE)	E-PGB-L32E2 (FUSE	CEDM NORM ACU FAN B M-HCN-A02B
E-PGB-L34D2 (FUSE)	E-PGB-L34D2 (FUSE)	CTMT NORM ACU FAN D M-HCN-A01D

THE ABOVE BACKUP PROTECTION CONTROL CIRCUIT FUSES WERE ADDED BY DCP NO. 1SE-PK-029, 2CE-PK-029, 3CE-PK-029

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a MINIMUM ~~total~~ feedwater flow of ~~987~~ 750 gpm at a pressure of 1260 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ~~with the~~ ENSURES THAT A minimum WATER volume OF 300,000 GALLONS ~~ensures that sufficient water~~ is available to maintain the Reactor Coolant System at HOT STANDBY for ~~24~~ 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the nitrogen accumulator at a pressure ≥ 400 psig is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV which holds the plant at hot standby while dissipating core decay heat or which allows a flow of sufficient steam to maintain a controlled reactor cooldown rate. A pressure of 400 psig retains sufficient nitrogen volume for 4 hours of operation at hot standby plus 6.5 hours of operation to reach cold shutdown under natural circulation conditions in the event of failure of the normal control air system.

PROOF AND REVIEW

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to ~~(90)~~°F and 120°F ~~(275)~~ 230 psig are based on a steam generator RT_{NDT} of ~~(30)~~ 40°F and are sufficient to prevent brittle fracture.

3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the essential cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

The OPERABILITY of the essential spray pond system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 27-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the essential chilled water system ensures that sufficient cooling capacity is available for continued operation of equipment and control room habitability during accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

The OPERABILITY of the control room essential filtration system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

The OPERABILITY of the ESF pump room air exhaust cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with 10 detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 20 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Review Board. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommedations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may

SNUBBERS (Continued)

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.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected area(s) until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tank~~s~~ by verifying either the weight or the level of the tanks. ~~Level measurements are made by either a U.L. or F.M. approved method.~~ CO_2

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

PROOF AND REVIEW

3/4.7.12 FIRE-RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.13 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

PROOF AND REVIEW

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.010 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

PROOF AND REVIEW

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The OPERABILITY of the motor-operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The surveillance requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) manipulator cranes will be used for movement of CEAs and fuel assemblies, (2) each crane has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulate RCS at a flow rate equal to or greater than 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (23 feet) is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

A nominal

3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

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3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue PHYSICS TESTING.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth, (2) determine the reactor stability index and damping factor under xenon oscillation conditions, (3) determine power distributions for non-normal CEA configurations, (4) measure rod shadowing factors, and (5) measure temperature and power coefficients. Special test exception permits MTC to exceed limits in Specification 3.1.1.3 during performance of PHYSICS TESTS.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

This special test exception permits the CEAs to be positioned beyond the insertion limits and reactor coolant cold leg temperature to be outside limits during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 MINIMUM TEMPERATURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{cold} below the minimum critical temperature during PHYSICS TESTS which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions. The Low Power Physics Testing Program at low temperature (300°F) is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. Regulating CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

3/4.10.7 SPENT FUEL POOL LEVEL

This special test exception permits loading of the initial core with the spent fuel pool dry.

3/4.10.8 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, ~~300~~°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

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3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGE TO ONSITE EVAPORATION PONDS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that at any time during the life of the nuclear station, the annual total body dose due to ground contamination of an UNRESTRICTED AREA, arising from transportation and deposition by wind of the accumulated activity discharged to the pond from the secondary system of the plant (if the pond gets dried up) on the UNRESTRICTED AREA, is within the guidelines of 10 CFR Part 20 for the above-mentioned postulated event.

Restricting the concentrations of the secondary liquid wastes discharged to the onsite evaporation ponds will restrict the quantity of radioactive material that can get accumulated in the ponds. This, in turn, provides assurance that in the event of an uncontrolled release of the pond's contents to an UNRESTRICTED AREA, the resulting total body annual exposure from ground contamination to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will be within 0.5 rem.

This specification applies to the secondary system liquid waste discharges of radioactive materials from all reactor units to the onsite evaporation ponds. Since the chemical neutralizer tank concentrations will bound concentrations in other secondary waste discharges, surveillance requirements stipulate that sampling and analysis of other secondary waste discharges need be performed only if the sampling and analysis of the contents of the chemical neutralizer tank shows that the neutralizer tank concentration exceeds the specified LLD.

The required detection capabilities for radioactive materials in the secondary liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. ~~The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with~~

Delete

~~the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.~~

~~This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.~~ Delete

3/4.11.1.3 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

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DOSE RATE (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from all reactor units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

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3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the GASEOUS RADWASTE SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification ~~implements~~ ADDRESSES the requirements of ~~10 CFR Part 50.36a and~~ General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if

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TOTAL DOSE (Continued)

the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

PROOF AND REVIEW

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

SITE AND EXCLUSION BOUNDARIES

5.1.1 The site and exclusion boundaries shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

GASEOUS RELEASE POINTS

5.1.3 The gaseous release points shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, prestressed concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 146 feet.

b. Nominal inside height = 206.5 feet.

c. Minimum thickness of concrete walls = 3 feet, 8 inches.

d. Minimum thickness of concrete roof = 3 feet, 8 inches.

e. Minimum thickness of concrete floor pad = 10.5 feet.

f. Nominal thickness of steel liner = 0.25 inch.

g. Nominal net free volume = 2.6×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 300°F.

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The main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes, a turbine trip (without reactor trip or cutback) from RATED THERMAL POWER with a coincident loss of condenser heat sink (i.e., no steam bypass) is assumed. The combined relieving capacity of the pressurizer safety valves, and the heat removal capacity of the MSSVs, is sufficient to maintain the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks and 110% of design pressure for all other events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity of all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity of 19.53×10^6 lbm/hr given in Table 3.7-1 as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainity) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity of 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

Although the variable high power reactor trip is not relied on for the limiting overpressure events, the ceiling on this trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

SP = Allowable Power Level + 9.8

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is the ratio of the available relieving capacity of the total steam flow at rated power

10 = total number of main steam safety valves for one steam generator

N = number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves

109.2 = ratio of main steam safety valve relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainity (see above text)

9.8 = BAND between the maximum thermal power and the variable overpower trip setpoint ceiling

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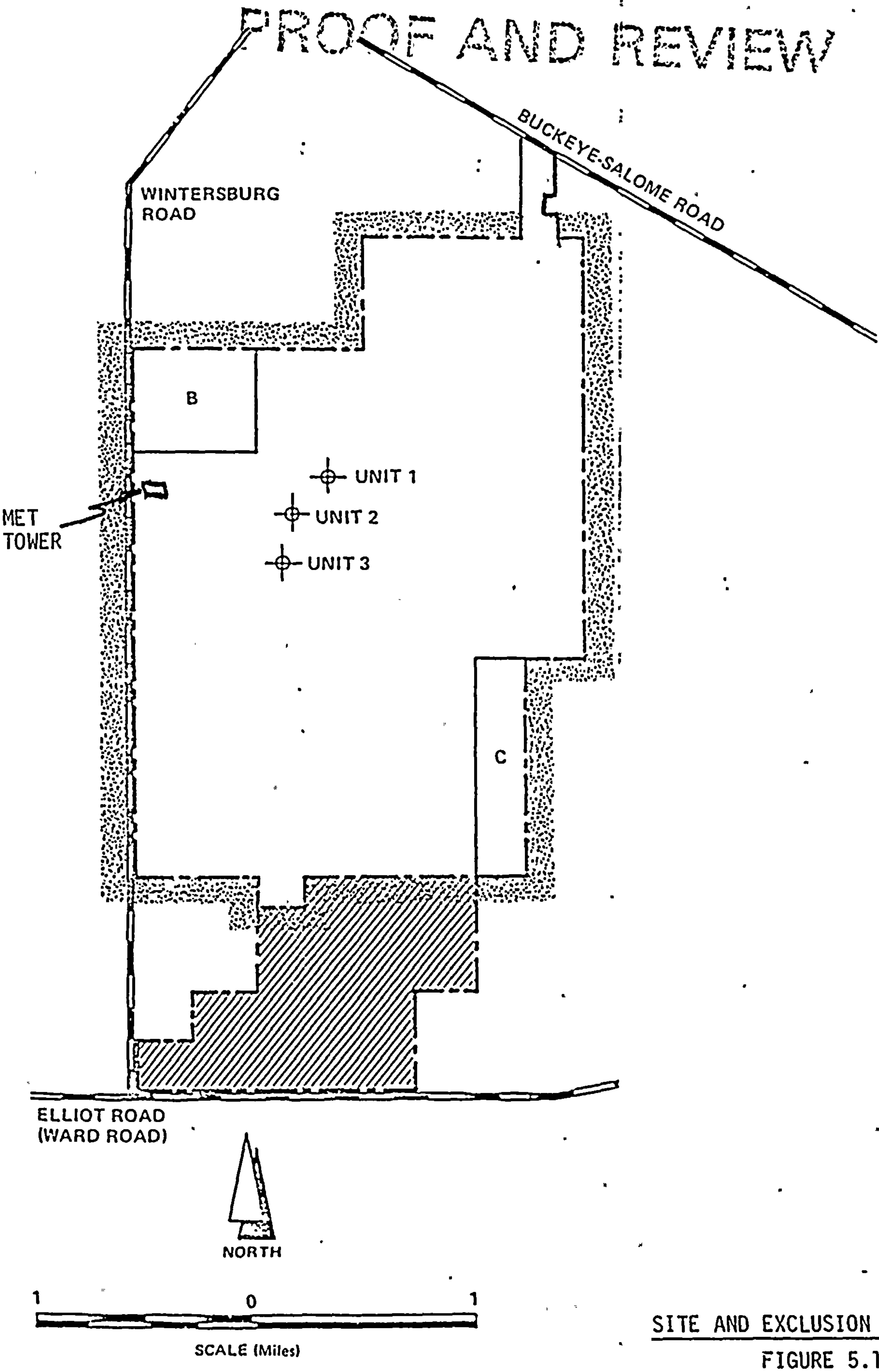
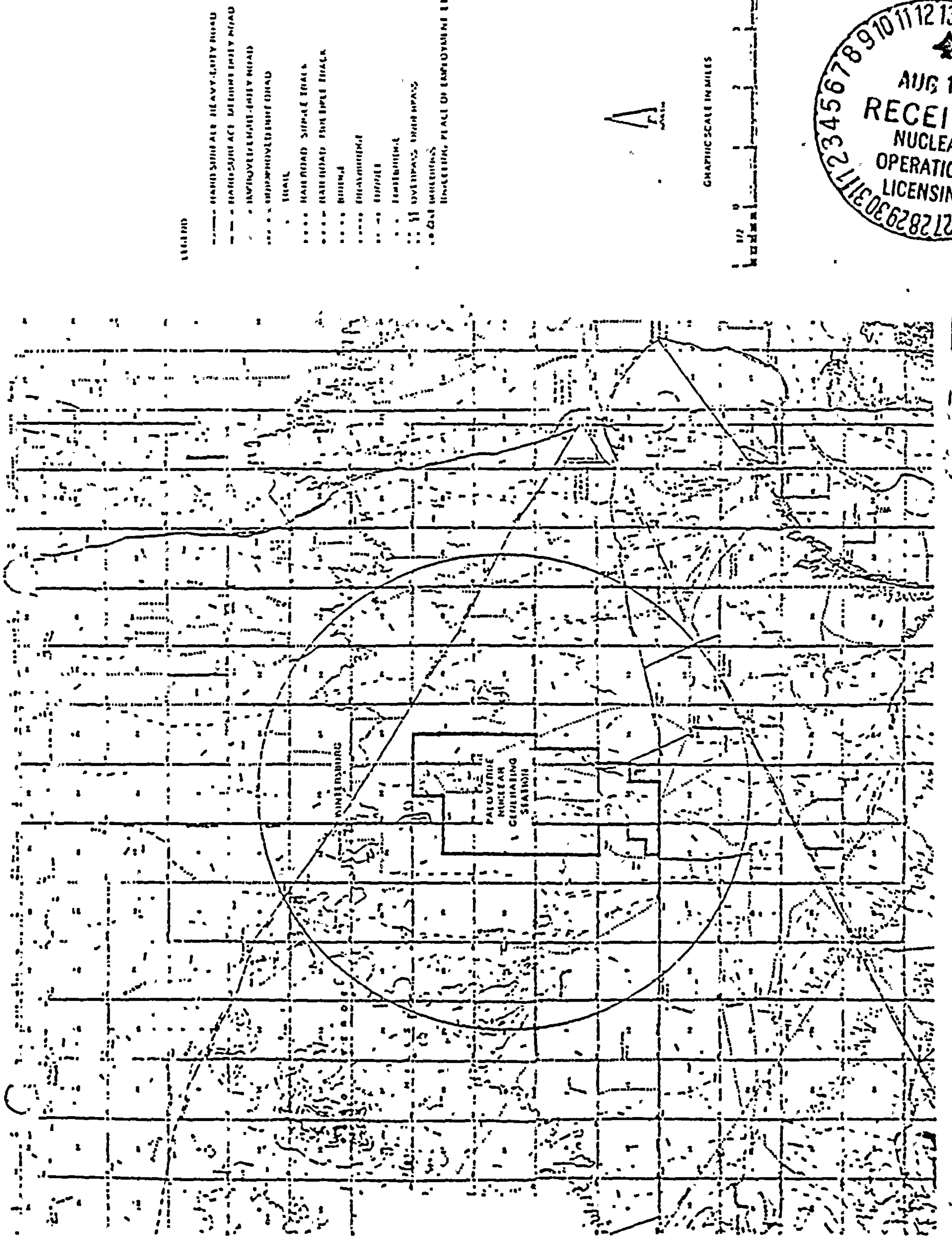


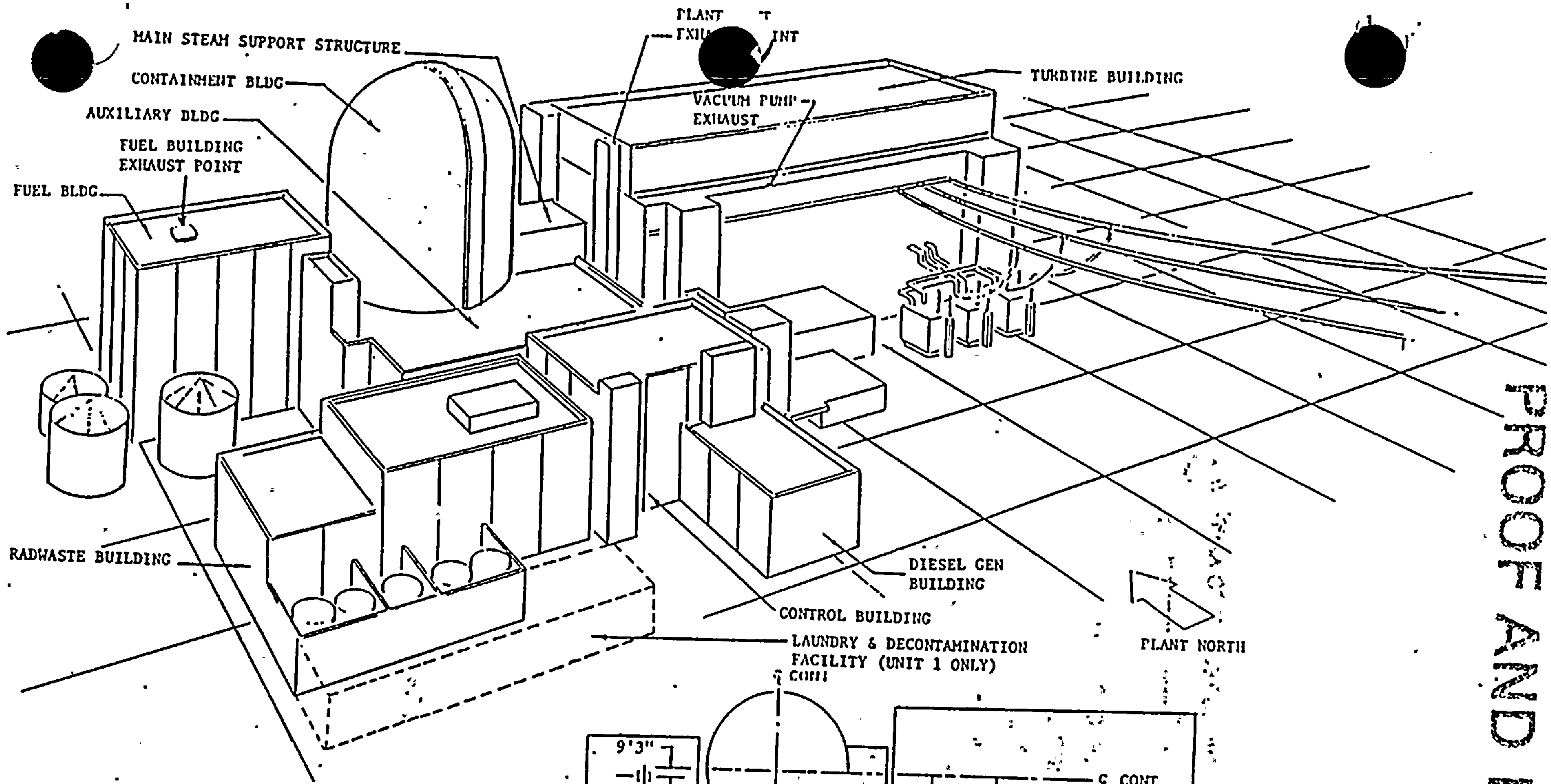
SITE AND EXCLUSION BOUNDARIES

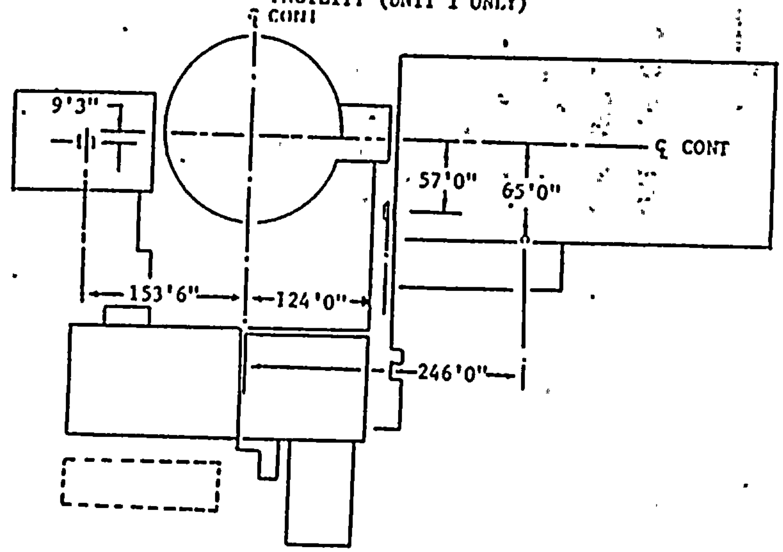
FIGURE 5.1-1



LOW POPULATION ZONE
PALO VERDE NUCLEAR GENERATING STATION

FIGURE 5.1-2





EXHAUST POINTS KEY PLAN

FIGURE 5.1-3

GASEOUS RELEASE POINTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 241 fuel assemblies with each fuel assembly containing 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately 1950 grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 76 full-length and 13 part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable surveillance requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,900 + 300/-0 cubic at a nominal T_{avg} of 593°F.

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5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal ~~9.43~~ 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1[illegible] fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

TABLE

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - ~~T_{avg}~~ Temperature from ≤ ~~200~~ 70°F to ≥ 565°F; cooldown cycle - ~~T_{avg}~~ Temperature from ≥ 565°F to ≤ ~~200~~ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heatup cycle - Pressurizer temperature from ≤ ~~200~~ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ ~~200~~ 70°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature between 100°F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one-half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle.
	15000 POWER CHANGE CYCLES	CYCLES FROM 15% TO 100% FULL LOAD, AT A RATE OF 5% PER MINUTE EITHER INCREASING OR DECREASING (30,000 CYCLES TOTAL)
	10^6 STEP CHANGES OF 100 PSI AND 10°F (20°F FOR SURGE LINE)	PRESSURE VARIATIONS BETWEEN THE PRESSURIZER PRESSURE SETPOINT FOR BACKUP HEATER ACTUATION AND SPRAY VALVE OPENING. TEMPERATURE VARIATIONS DUE TO CEA CONTROLLER; 2000 STEP CHANGE OF 10% FULL POWER.
	200 PRIMARY SYSTEM LEAK TEST CYCLES	LEAK TEST PRIMARY SYSTEM AT A PRESSURE OF 2250 PSI AT A TEMPERATURE FROM 120°F TO 400°F

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - ~~T_{avg}~~ TEMPERATURE from ≤ ~~200~~ 70°F to ≥ 565°F; cooldown cycle - ~~T_{avg}~~ TEMPERATURE from ≥ 565°F to ≤ ~~200~~ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heatup cycle - Pressurizer temperature from ≤ ~~200~~ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ ~~200~~ 70°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psig with RCS temperature between 120 °F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle.
	15,000 power change cycles	Cycles from 15% to 100% full load, at a rate of 5% per minute, either increasing or decreasing. (30,000 cycles total)
	10^6 step changes of 100 psi and 10°F (20°F for surge line)	Pressure variations between the pressurizer pressure setpoint for backup heater actuation and spray valve opening. Temperature variations due to CEA controller; 2000 step change of 10% full power.
	200 PRIMARY SYSTEM LEAK TEST CYCLES	LEAK TEST PRIMARY SYSTEM AT A PRESSURE OF 2250 PSI AT A TEMPERATURE FROM 120 °F TO 400 °F.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray Nozzle	Unlimited number of cycles ~~Calculate usage factor per Table 5.7-2~~.	Main spray (4 pumps operating) Main spray (less than four RCP operating) with fluid $\Delta T_m > 200°F$.
	Calculate usage factor per Table 5.7-2	Auxiliary spray ~~with fluid $\Delta T_a > 200°F$~~. at various initial fluid temperatures with fluid $\Delta T_a \leq 200°F$
		Auxiliary spray with fluid $\Delta T_a > 200°F$

ΔT_m = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

ΔT_a = The difference in temperature between the pressurizer and auxiliary spray water as adjusted by the instrument correction factor.

TABLE 5.7-1 (Cont'd)

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Pressurizer Spray Nozzle		Main Spray (4 pumps operating)
	Unlimited number of cycles	Main Spray (Less than 4 pumps operating) - with fluid $\Delta T_m \le \cancel{X}$°F. 200
		Auxiliary spray at various initial fluid temperatures with fluid $\Delta T_a \le \cancel{X}$°F. 200
	Calculate usage factor per Table 5.7-2	Main spray (less than 4 pumps operating) with fluid $\Delta T_m > \cancel{X}$°F. 200
		Auxiliary spray with fluid $\Delta T_a > \cancel{X}$°F. 200

ΔT_m = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

ΔT_a = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.

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PRESSURIZER SPRAY NOZZLE USAGE FACTOR

Main Spray				Auxiliary Spray			
ΔT_m	N_A	N	N/N_A	ΔT_a	N_A	N	N/N_A
201-250	7900			201-250	50000		
251-300	4500			251-300	2200		
301-350	2900			301-350	1300		
351-400	1900			351-400	850		
401-450	1200			401-450	550		
451-500	850			451-500	375		
501-550	555			501-550	225		
				551-600	150		
		$\Sigma N/N_A =$	________			$\Sigma N/N_A =$	________

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ________________

$\Sigma N/N_A$ (Aux. Spray) ________________

Total ________________ = Cumulative Usage Factor

TABLE 5.7 (Continued)

Where:

$\Delta T_a = (T_{101} - T_{229}) + 60$

$\Delta T_m = (T_{101} - T_{103*} \text{ or } {}_{104*}) + 70$

NA = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow throughout the pressurizer spray nozzle.

2. If the difference between pressurizer water temperature and the spray water temperature exceeds 200°F each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N by N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

4. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of nozzle fatigue shall be performed within 90 days. The evaluation shall determine that the nozzle remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated.

*Use lower of two temperatures.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Director of Nuclear Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated SRO, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1. or as specified in the FSAR.

UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 or as specified in the FSAR and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the ~~(2)~~ other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. 3

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

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*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (THIS EXCLUDES THE STA WORKING HOURS):

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the Director of Nuclear Operations or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Director of Nuclear Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

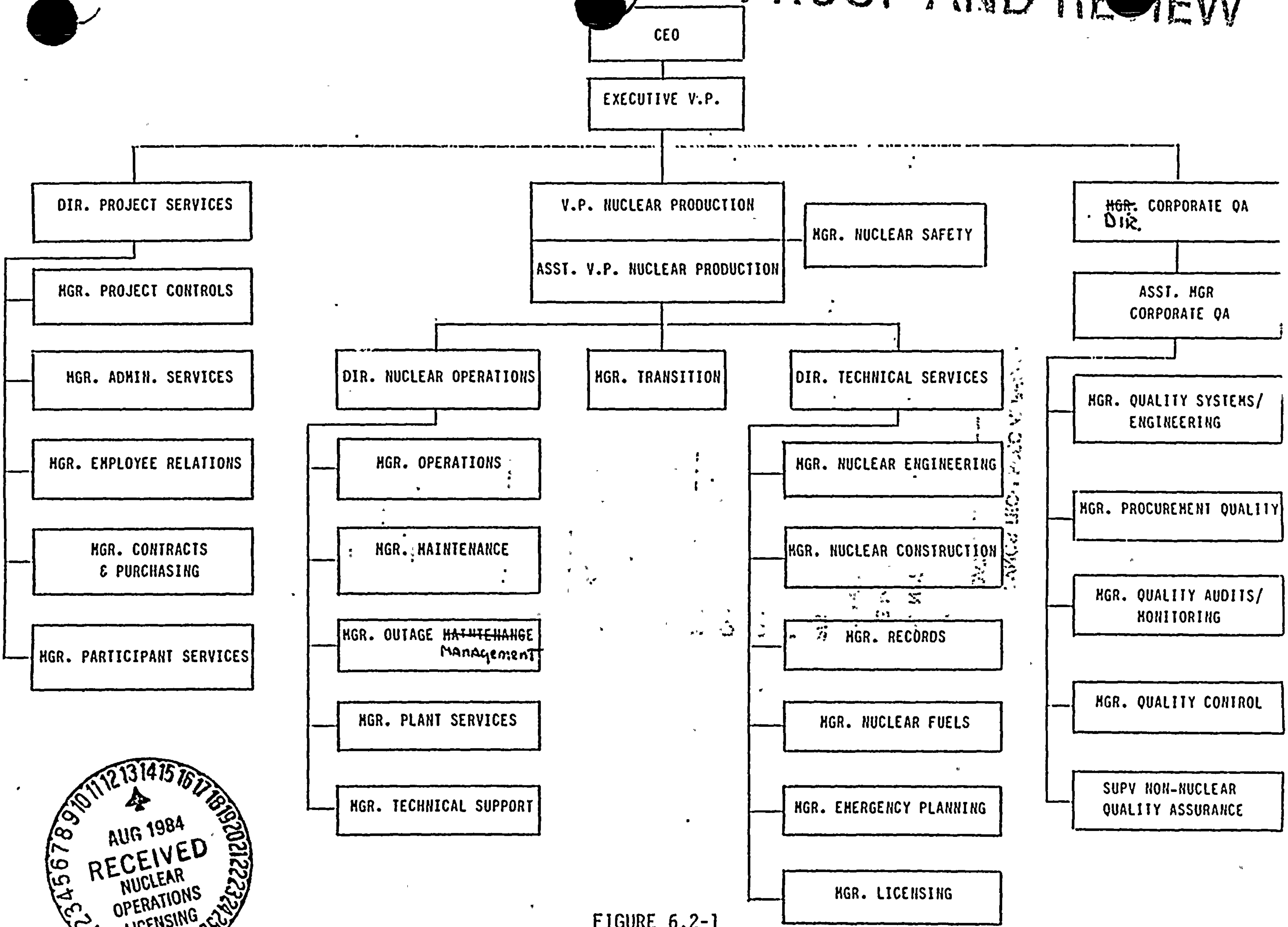

FIGURE 6.2-1
OFFSITE ORGANIZATION

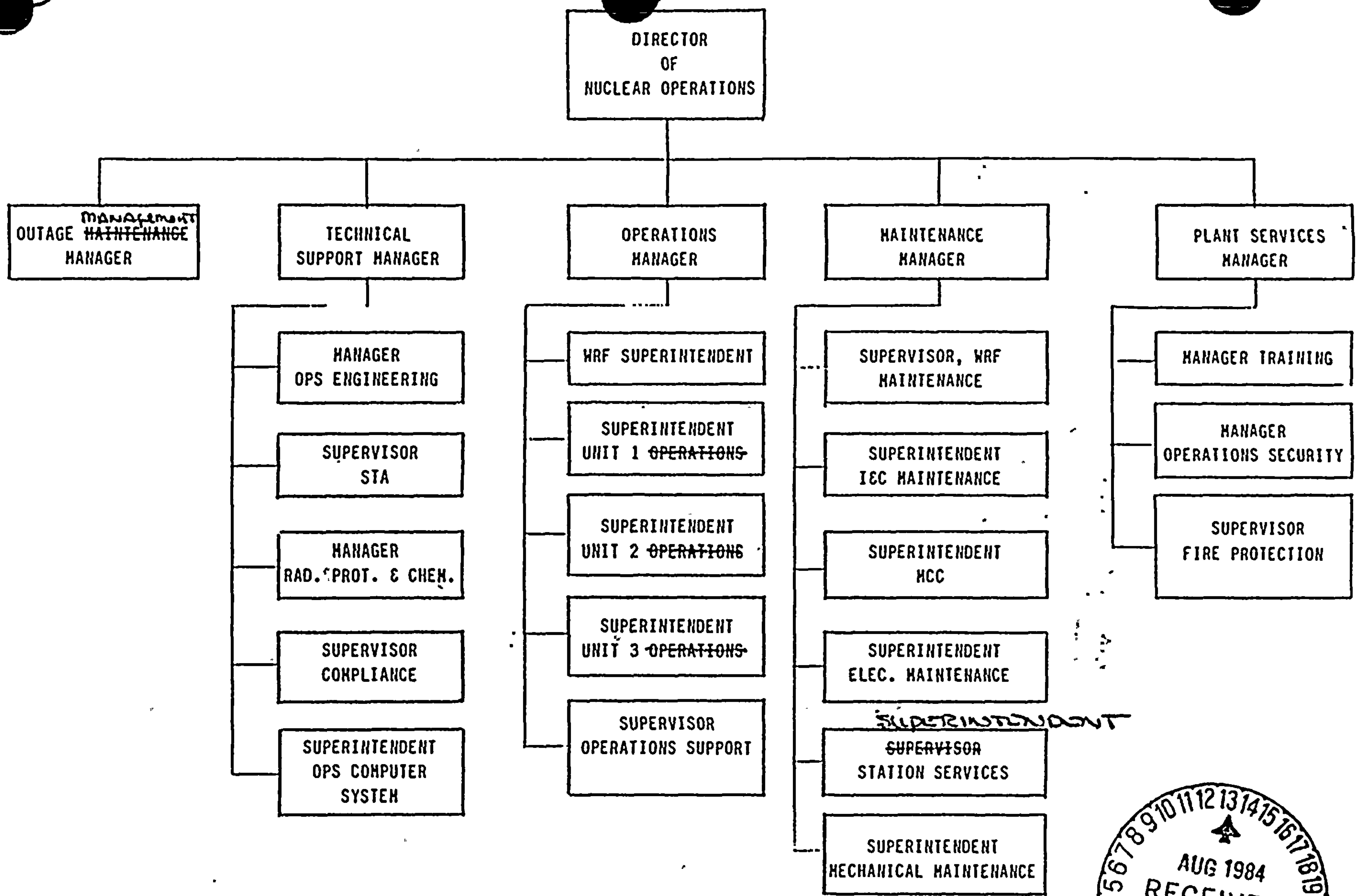

FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 ~~The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.~~

STET
See Attachment 6-A

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, Director of Nuclear Operations, and the Supervisor, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety. ~~and Supervisor of the NSG.~~

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978, as endorsed by Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

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*Not responsible for sign-off function.

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6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director of Nuclear Operations or his designee and shall meet or exceed the requirements and recommendations of Section 5.5 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees; and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Director of Nuclear Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

Member:	Technical Support Manager
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Plant Services Manager
Member:	Engineering Manager
Member:	U1, U-2, U-3 ~~Operations~~ Superintendents
Member:	STA Supervisor
Member:	~~Training Manager~~ I AND C Superintendent
Member:	Radiation Protection and Chemistry Manager
Member:	Quality Systems/Engineering Manager

The Director of Nuclear Operations shall designate the Chairman and Vice-Chairman in writing.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairman, or his designated alternate.

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MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairman, or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice Chairman, or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Board shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the D.N.O.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

f. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

g. Review of all Administrative Control Procedures and changes.

h. Review of all proposed changes to Appendix "A" Technical Specifications.

AUTHORITY

6.5.1.7 The Plant Review Board (PRB) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Vice President-Nuclear Production, DNO and NSG of disagreement between PRB and the DNO; however, the DNO shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Nuclear Safety Group.

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Director Nuclear Operations (DNO) shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Manager of Technical Support or the Manger of Engineering as previously designated by the Director of Nuclear Ops. Test results shall be approved by the Director of Nuclear Operations or the Manager of Technical Support.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the DNO; or by the Manager, Technical Support as previously designated by the DNO.

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the DNO to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the DNO, the Manager Technical Support, the Manager Operations, or the Manager Maintenance.

6.5.2.6 Review of the station security program, and implementing procedures, and submittal of recommended changes shall be approved by the DNO and transmitted to the Vice President-Nuclear Production and to the NSG.

ACTIVITIES (Continued)

6.5.2.7 Review of the station emergency plan, and implementing procedures, and submittal of recommended changes shall be approved.

6.5.2.8 The DNO shall assure the performance of a review by a qualified individual/organizaiton of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence.

6.5.2.9 The DNO shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

REVIEW

6.5.3.4 The NSG shall be responsible for the review of:

a. The safety evaluations program and its implementation for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question; this review will be done by the audit of selected safety evaluations.

b. Proposed changes to procedures, equipment, or systems which an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications of this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect neclear safety;

g. ALL REPORTABLE EVENTS Requiring 24 hour notification

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the congnizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

AUDITS (Continued)

e. Any other area of unit operation considered appropriate by the Manager Nuclear Safety or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. an outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

i. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Manager of Nuclear Safety on those areas of responsibility specified in Specifications 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews shall be prepared monthly for the Manger, Nuclear Safety, who will distribute it to the Vice President-Nuclear Production, Director of Nuclear Operations, and to the management positions responsible for the areas reviewed.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS: Requiring 24 hour notification.

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted ot the Manager of Nuclear Safety and the Vice President-Nuclear-Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production, and Director of Nuclear Operations and Manager Nuclear Safety shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared . The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Manager Nuclear Safety and the Vice President-Nuclear Production within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirments of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC Addressable Constants based primarily on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

PROOF AND REVIEW

PROCEDURES AND PROGRAMS (Continued)

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each program or procedure of specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5:1 and approved prior to implementation. Programs and administrative control procedures shall be approved by the Director of Nuclear Operations, or designated alternate. Implementing procedures shall be approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom is the Shift Supervisor or Assistant Shift Supervisor on the unit affected.

c. The change is documented, reviewed in accordance with Specification 6.5.1 and approved by the Director of Nuclear Operations or congnizant department head, as designated by the Director of Nuclear Operations, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the congnizance of the NSG at least once per 24 months:

a. PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

A program to reduce leakage from those portions of systems outside ocntainment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

PROCEDURES AND PROGRAMS Continued)

b. In-Plant Radiation Monitoring

A program will be written and issued prior to system being declared operational which will ensure the capability to accurately determine the airborne iodine concnetration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment. Operational

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required ot initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

PROCEDURES AND PROGRAMS (Continued)

e. Post-accident Sampling

A Program will be written and issued prior to system being declared operational which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The intial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,**e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 805 of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT***

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

***A single submittal may be made for a multiple unit station.

PROOF AND REVI...

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*** (Continued)

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted with 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

***A single submittal may be made for a multiple unit station.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive gaseous effluents released from the unit or station during the previous calendar year. All assumptions used in making these assessments, i.e. specific activity, exposure time, location and meteorological models shall be included in these. The assessment of radiological doses shall be performed in accordance with methodology and parameters in the ODCM. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurment or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

**In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde),

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least for the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

6.10 RECORD RETENTION (Continued)

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leaks tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit desing modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure fo all individuals entering radiation control areas.

d. Records of gaseous radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Record of reactor tests and experiments.

g. Records of training and qualification for current members unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of test and experiments pursuant to 10 CFR 50.59.

k. Records of PRB meetings.

l. Records of the service lives of all hyyraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of specifications 6.5.2.8 and 6.8.4.

6.10 RECORD RETENTION (Continued)

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

6.11. RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requireing issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

PROOF AND REVIEW

6.12 HIGH RADIATION AREA (Continued)

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequancy specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, area accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorizied entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area. For individual areas accessible to personnel with radiation levels such that major portion of the body could recieve in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection pocedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP

a. Shall be submitted to the Commission in the Semiannual Radiaoactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

(1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

*Measurement made at 18 inches from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP) (Continued)

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reeliability of dose calculations or setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS* (Continued)

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems,

4) An evaluation of the change, which shows the predicted releases of radioactive materials in gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the licenses application and amendments thereto:

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.

b) Shall become effective upon review and acceptance by the PRB.

*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

ATTACHMENT B

PROOF AND REVIEW

PG VI

CHANGE TO:

JUSTIFICATION:

Typo's in page numbers.

PG VII

CHANGE TO:

JUSTIFICATION:

Typo's in page numbers.

3/4 VIII

DELETE:

JUSTIFICATION:

APS doesn't have Halon Systems and do not have the Tech Spec.

CHANGE:

JUSTIFICATION:

Type in page number.

PG IX

CHANGE:

JUSTIFICATION:

Typo's in page numbers.

3269D/0126D

PROOF AND REVIEW

PG XVI CHANGE:

Meeting Frequency 6-7

JUSTIFICATION:

Typo's in page numbers.

PG XIX CHANGE:

3.4-1 ...Power with the primary...

JUSTIFICATION:

Typo.

PG XXI CHANGE:

PG XXII

JUSTIFICATION:

Typo's.

PG 1-2 CHANGE 1.7.9.2 TO:

...provided in table 3.6-0 of specification 3.6.1.1

JUSTIFICATION:

To reference the proper correct table and specification number.

PG 1-7 ADD 1.3.9

Fire Protection Evaluation Report as discussed in the PVNGS FSAR.

JUSTIFICATION:

Tech Spec 3.3.3.7 refers to a fire detection zone. PVNGS uses the FPER zone terminology instead of a fire detection zone. This change is consistent with PVNGS Operations Terminology and FSAR.

PROOF AND REVIEW

PG 1-9 Add # note to startup

JUSTIFICATION:

PVNGS Operations and engineering personnel needs this change to clarify and locate the special test exception for initial criticality and low power physics testing. Since this condition is not "Normal" Operating Procedure when the operations go critical below 350°F they may try to take some kind of corrective action to comply with Tech Specs not realizing that they are operating in a special test exception area. This change will assist Resident Inspectors identify and locate a Special Test Exception Factor.

PG 2-3 TABLE 2.2-1 ITEM 1.A.7.c

Change allowable value to 42.1%.

JUSTIFICATION:

New CE number.

PG 2-5 ADD to FOOTNOTE 6

Setpoints are % of 100% power flow conditions.

JUSTIFICATION:

CE words to clarify and explain what exactly is happening.

PG B2-4 ADD TO STEAM GENERATOR LEVEL - LOW

...10 minutes before auxiliary feedwater is required to prevent degraded core cooling

JUSTIFICATION:

PVNGS uses the term auxiliary vs emergency emergency as shown in the Tech Spec. Also the added phrase provides more detailed description as to what is really happening.

PG B2-5 CHANGE DNBR-LOW

...Floor of 1861 psia.

JUSTIFICATION:

CE changed number

PROOF AND REVIEW

PG B2-6 **CHANGE:**

See page.

JUSTIFICATION:

CE changed numbers.
New descriptions and additions for Steam Generator Level High and Reactor Coolant Flow - Low provide more accurate and detailed information to assist the Operating Personnel.

PG B2-7 **CHANGE:**

See page.

JUSTIFICATION:

Again, to provide a more accurate statement for CPC Addressable Constants.

PG 3/4 1-1 **CHANGE ACTION STATEMENT**

See page.

JUSTIFICATION:

This change has been continuously discussed with the NRC. As the spec is presently written there is alot of operator confusion as to if the operator immediately initiates boration or goes to Spec. 3.1.3.6 which allows the operator to have two hours to take corrective action Many of our Operations People have some confusion which will lead them to taking various corrective actions. The proof and review LCO 3.1.1.1 is in conflict with LCO 3.1.3.6. The action statement in LCO 3.1.1.1 requires immediate boration to reestablish 6% delta K/K shutdown margin, if violated. However, LCO 3.1.3.6 allows two (2) hours to recover rod position above the Transient Insertion Limit of Figure 3.1.3 or Figure 3.1.4. Since Transient Insertion Limits on these curves represents a value of available shutdown margin equivalent to 6% delta K/K, this means there are two different action requirements for exceeding the same limit. In actuality, the only verification of shutdown margin that is required to be performed during normal critical operation is the CEA position verification.