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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
License No. DPR-46

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

   A. The applications for amendment by Nebraska Public Power District dated June 10, 1982, November 24, 1982 and February 25, 1983 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 82, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 29, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 82

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Revise Appendix A as follows:

| Remove | Insert |
|---|---|
| iii | iii |
| iiia | iiia |
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| 61 | 61 |
| 137a | 137a |
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| 235a | - |

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SAFETY LIMITS

## 1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications

A. Reactor Pressure >800 psia and Core Flow >10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety.

B. Core Thermal Power Limit (Reactor Pressure <800 psia and/or Core Flow <10%)

When the reactor pressure is $\leq$800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

LIMITING SAFETY SYSTEM SETTINGS

## 2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66\ W + 54\%$

where:

S = Setting in percent of rated thermal power (2381 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% coreflow at 100% power)

SAFETY LIMIT

## 1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications:

1. The reactor vessel dome pressure shall not exceed 1337 psig at any time when irradiated fuel is present in the reactor vessel.

2. The reactor vessel dome pressure shall not exceed 75 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

LIMITING SAFETY SYSTEM SETTING

## 2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications:

1. The limiting safety system settings shall be as specified below:

Protective Action/Limiting Safety System Setting

A. Scram on Reactor Vessel high pressure-

$\leq$1045 psig

B. Relief valve settings-

1080 psig $\pm$ 11 psi (2 valves)
1090 psig $\pm$ 11 psi (3 valves)
1100 psig $\pm$ 11 psi (3 valves)

C. Safety valve settings-

1240 psig $\pm$ 13 psi (3 valves)

2. Action shall be taken to decrease the reactor vessel dome pressure below 75 psig or the shutdown cooling isolation valves shall be closed.

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TABLE 3.1.1

REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

| Reactor Protection System Trip Function | Applicability Conditions Mode Switch Position: Shutdown | Startup | Refuel | Run | Trip Level Setting | Minimum Number of Operable Channels Per Trip Systems (1) | Action Required When Equipment Operability is Not Assured (1) |
|---|---|---|---|---|---|---|---|
| Mode Switch in Shutdown | X(7) | X | X | X | | 1 | A |
| Manual Scram | X(7) | X | X | X | | 1 | A |
| IRM (17) High Flux | X(7) | X | X | (5) | ≤ 120/125 of indicated scale | 3 | A |
| Inoperative | | X | X | (5) | | 3 | A |
| APRM (17) High Flux (Flow biased) | | | | X | ≤ (0.66W+54%) (14) $\frac{FRP}{MFLPD}$ | 2 | A or C |
| High Flux | X(7) | X(9) | X(9) | (16) | ≤ 15% Rated Power | | A or C |
| Inoperative | | X(9) | X(9) | X | (13) | 2 | A or C |
| Downscale | | (11) | | X(12) | > 2.5% of indicated scale | 2 | A or C |
| High Reactor Pressure NBI-PS-55 A,B,C, & D | | X(9) | X(10) | X | ≤ 1045 psig | 2 | A |
| High Drywell Pressure PC-PS-12 A,B,C, & D | | X(9)(8) | X(8) | X | ≤ 2 psig | 2 | A or D |
| Reactor Low Water Level NBI-LIS-101 A,B,C, & D | | X | X | X | ≥ + 12.5 in. indicated level | 2 | A or D |
| Scram Discharge Instrument Volume High Water Level CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D | | X | X(2) | X | ≤ 92 inches | 3 (18) | A |

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the mode switch is not in RUN.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 11 operable LPRM detectors to an APRM.

14. W is the recirculation flow in percent of rated flow.

15. This note deleted.

16. The 15% APRM scram is bypassed in the RUN mode.

17. The APRM and IRM instrument channels function in both the Reactor Protection System and Reactor Manual Control System (Control Rod Withdraw Block, Section 3.2.C.). A failure of one channel will affect both of these systems.

18. The minimum number operable associated with the Scram Discharge Instrument Volume are three instruments per Scram Discharge Instrument Volume and three level devices per RPS channel.

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TABLE 4.1.1 (Page 2)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

| Instrument Channel | Group (2) | Functional Test | Minimum Frequency (3) |
|---|---|---|---|
| High Water Level in Scram Discharge Instrument Volume CRD LS-231 A & B<br>CRD-LS-234 A & B<br>CRD-LT-231 C & D<br>CRD-LT-234 C & D | A | Trip Channel and Alarm | Once/3 Months |
| Main Steam Line High Radiation<br>RMP-RM-251 A,B,C, & D | B | Trip Channel and Alarm (4) | Once/Week |
| Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D<br>MS-LMS-80 A,B,C, & D | A | Trip Channel and Alarm | Once/Month (1) |
| Turbine Control Valve Fast Closure<br>TGF-63/OPC -1,2,3,4 | A | Trip Channel and Alarm | Once/Month (1) |
| Turbine First Stage Pressure Permissive MS-PS-14 A,B,C, & D | A | Trip Channel and Alarm | Once/3 Months |
| Turbine Stop Valve Closure<br>SVOS-1 (1), SVOS-1 (2)<br>SVOS-2 (1), SVOS-2 (2) | A | Trip Channel and Alarm | Once/Month (1) |
| Reactor Pressure Permissive<br>NBI-PS-51 A,B,C & D | A | Trip Channel and Alarm | Once/3 Months |

3.1 BASES (cont'd.)

against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 36 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, diverse indication (two level switches and two level transmitters for each discharge volume) has been provided in the instrument volumes which alarm and scram the reactor when the volume of water reaches 92 inches. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph VII.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (refer-

4.1 BASES (cont'd.)

revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than $20 \times 10^{-6}$ failures/hour. The bi-stable trip circuits are predicted to have an unsafe failure rate of less than $2 \times 10^{-6}$ failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a

TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

| Function | Trip Level Setting | Minimum Number Of Operable Instrument Channels/Trip System(5) |
|---|---|---|
| APRM Upscale (Flow Bias) | $\leq (0.66W + 42\%)\ \frac{FRP}{MFLPD}$ (2) | 2(1) |
| APRM Upscale (Startup) | $\leq$ 12% | 2(1) |
| APRM Downscale (9) | $\geq$ 2.5% | 2(1) |
| APRM Inoperative | (10b) | 2(1) |
| RBM Upscale (Flow Bias) | $\leq (0.66W + 40\%)$ (2) | 1 |
| RBM Downscale (9) | $\geq$ 2.5% | 1 |
| RBM Inoperative | (10c) | 1 |
| IRM Upscale (8) | $\leq$ 108/125 of Full Scale | 3(1) |
| IRM Downscale (3)(8) | $\geq$ 2.5% | 3(1) |
| IRM Detector Not Full In (8) | | 3(1) |
| IRM Inoperative (8) | (10a) | 3(1) |
| SRM Upscale (8) | $\leq 1 \times 10^5$ Counts/Second | 1(1)(6) |
| SRM Detector Not Full In (4)(8) | ($\geq$ 100 cps) | 1(1)(6) |
| SRM Inoperative (8) | (10a) | 1(1)(6) |
| Flow Bias Comparator | $\leq$ 10% Difference In Recirc. Flows | 1 |
| Flow Bias Upscale/Inop. | $\leq$ 110% Recirc. Flow | 1 |
| SRM Downscale (8)(7) | $\geq$ 3 Counts/Second (11) | 1(1)(6) |
| SDV Water Level High CRD-231F, 234E | $\leq$ 46 inches | 1(12) |

3.6.H Shock Suppressors (Snubbers)

1. During all modes of operation except Cold Shutdown and Refuel, all safety related snubbers shall be operable except as noted in 3.6.H.2 through 3.6.H.5 below.

2. The snubbers listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 are required to protect the primary coolant system or other safety related systems or components. All others are therefore exempt from these specifications.

3. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.6.H.4 on the supported component or declare the supported system or subsystem inoperable and follow the appropriate ACTION statement for that system.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to, removed, or substituted for, by analysis, from safety related systems without prior License Amendment to Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4, provided that a revision to these tables is included with a subsequent License Amendment request.

4.6.H Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4.

1. All snubbers shall be visually inspected in accordance with the following schedule:

| Number of Snubbers Found Inoperable During Inspection or During Inspection Interval | Next Required Inspection Interval |
|---|---|
| 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% |

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation and by type, hydraulic or mechanical. These four groups may be inspected independently according to the above schedule.

2. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
| --- | --- |
| | 4.6.H Shock Suppressors (Snubbers) (cont'd)<br><br>structure 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, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.6.H.6 or 4.6.H.7 as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.<br><br>3. At least once per 18 months during shutdown, a representative sample, 10% of the total of each type of snubber in use in the plant, shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.6.H.5 or 4.6.H.6, an additional 10% of that type of snubber shall be functionally tested.<br><br>4. The representative sample selected for functional testing shall include various configuration, operating environments and the range of size and capacity of snubbers. Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 may be used jointly or separately as the basis for the sampling plan. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | 4.6.H Shock Suppressors (Snubbers) (cont'd)<br><br>Concurrent with the first in-service visual inspection and at least once per 18 months thereafter, the installation and maintenance records of each snubber listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records. |

Table 3.6.1

ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

| Snubber | Location |
|---|---|
| CS-SNUB-(CS-S1) | R-903-SE |
| CS-SNUB-(CS-S10) | R-931-NE |
| CS-SNUB-(CS-S11) | R-931-NE |
| CS-SNUB-(CS-S2) | R-903-SE |
| CS-SNUB-(CS-S3) | R-931-SE |
| CS-SNUB-(CS-S6) | R-881-SE QUAD |
| CS-SNUB-(CS-S7) | R-881-SE QUAD |
| CS-SNUB-(CS-VE7) | R-881-SE QUAD |
| HPCI-SNUB-(HP-S11) | R-859-HPCI RM |
| HPCI-SNUB-(HP-S15) | R-859-HPCI RM |
| HPCI-SNUB-(HP-S18) | R-859-SW QUAD |
| HPCI-SNUB-(HP-S18A) | R-859-HPCI RM |
| HPCI-SNUB-(HP-S22A) | R-859-HPCI RM |
| HPCI-SNUB-(HP-S4) | R-859-SW QUAD |
| HPCI-SNUB-(RF-S3) | R-859-HPCI RM |
| HPCI-SNUB-(RF-S4) | R-881-SW TORUS |
| HPCI-SNUB-(RF-S5) | R-881-SW TORUS |
| MS-SNUB-(BS-S1) | R-881-SW TORUS |
| MS-SNUB-(BS-S113A) | R-881-NW TORUS |
| MS-SNUB-(BS-S113B) | R-881-NW TORUS |
| MS-SNUB-(BS-S116A) | R-881-NW TORUS |
| MS-SNUB-(BS-S116B) | R-881-NW TORUS |
| MS-SNUB-(BS-S125A) | R-881-SW TORUS |
| MS-SNUB-(BS-S125B) | R-881-SW TORUS |
| MS-SNUB-(BS-S2) | R-881-SW TORUS |
| MS-SNUB-(BS-S3) | R-881-NW TORUS |
| MS-SNUB-(BS-S4) | R-903-A RHR HX RM |
| MS-SNUB-(BS-S5) | R-903-A RHR HX RM |
| MS-SNUB-(MS-S1) | R-859-HPCI RM |
| MS-SNUB-(MS-S10) | R-881-SW TORUS |
| MS-SNUB-(MS-S11) | R-881-SW TORUS |
| MS-SNUB-(MS-S111A) | R-903-A RHR HX RM |
| MS-SNUB-(MS-S11A) | R-881-SW TORUS |
| MS-SNUB-(MS-S12) | R-881-SW TORUS |
| MS-SNUB-(MS-S12A) | R-881-SW TORUS |
| MS-SNUB-(MS-S13) | R-903-B RHR HX RM |
| MS-SNUB-(MS-S13A) | R-903-B RHR HX RM |
| MS-SNUB-(MS-S13B) | R-903-B RHR HX RM |
| MS-SNUB-(MS-S14) | R-903-B RHR HX RM |
| MS-SNUB-(MS-S15) | R-931-B RHR HX RM |
| MS-SNUB-(MS-S15A) | R-931-B RHR HX RM |
| MS-SNUB-(MS-S16A) | R-881-NW TORUS |
| MS-SNUB-(MS-S16B) | R-881-NW TORUS |
| MS-SNUB-(MS-S17) | R-903-A RHR HX RM |
| MS-SNUB-(MS-S18) | R-903-A RHR HX RM |
| MS-SNUB-(MS-S19) | R-903-A RHR HX RM |
| MS-SNUB-(MS-S2) | R-859-HPCI RM |
| MS-SNUB-(MS-S20) | R-931-A RHR HX RM |
| MS-SNUB-(MS-S20A) | R-931-A RHR HX RM |
| MS-SNUB-(MS-S23) | R-881-NE TORUS |
| MS-SNUB-(MS-S24) | R-881-NE TORUS |
| MS-SNUB-(MS-S25) | R-859-NE QUAD |
| MS-SNUB-(MS-S26) | R-859-NE QUAD |

Table 3.6.1
ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

| Snubber | Location |
|---|---|
| MS-SNUB-(MS-S3) | R-859-HPCI RM |
| MS-SNUB-(MS-S4) | R-859-HPCI RM |
| MS-SNUB-(MS-S75) | R-931-A RHR HX RM |
| MS-SNUB-(MS-S76) | R-931-B RHR HX RM |
| MS-SNUB-(MS-S7A) | R-859-HPCI RM |
| MS-SNUB-(MS-S7B) | R-859-HPCI RM |
| MS-SNUB-(MS-S8) | R-881-SW TORUS |
| RCIC-SNUB-(RF-S1) | R-881-NE QUAD |
| RCIC-SNUB-(RF-S1A) | R-881-NE QUAD |
| RCIC-SNUB-(RF-S45C) | R-881-NE QUAD |
| RCIC-SNUB-(RF-S45D) | R-881-NE QUAD |
| RCIC-SNUB-(RF-S46A) | R-881-NE QUAD |
| RCIC-SNUB-(RF-S51A) | R-881-NE TORUS |
| RCIC-SNUB-(RF-S51B) | R-881-NE TORUS |
| REC-SNUB-(RCC-S20) | R-931-NW |
| REC-SNUB-(RCC-S21) | R-931-NW |
| REC-SNUB-(RCC-S22) | R-931-SW |
| REC-SNUB-(RCC-S3) | R-931-NE |
| REC-SNUB-(RCC-S4) | R-931-NE |
| RF-SNUB-(RF-S2) | R-881-NE TORUS |
| RF-SNUB-(RF-S6) | R-881-SE TORUS |
| RHR-SNUB-(RH-S103A) | R-859-SW QUAD |
| RHR-SNUB-(RH-S107A) | R-859-NW QUAD |
| RHR-SNUB-(RH-S20) | R-903-INJ V RM |
| RHR-SNUB-(RH-S21) | R-903-INJ V RM |
| RHR-SNUB-(RH-S22) | R-881-NW TORUS |
| RHR-SNUB-(RH-S23) | R-881-NW TORUS |
| RHR-SNUB-(RH-S24) | R-881-NW TORUS |
| RHR-SNUB-(RH-S25) | R-903-NW |
| RHR-SNUB-(RH-S25A) | R-903-NW |
| RHR-SNUB-(RH-S26) | R-903-NW |
| RHR-SNUB-(RH-S27A) | R-931-A RHR HX RM |
| RHR-SNUB-(RH-S29) | R-903-INJ V RM |
| RHR-SNUB-(RH-S30A) | R-881-SW TORUS |
| RHR-SNUB-(RH-S30B) | R-881-SW TORUS |
| RHR-SNUB-(RH-S32) | R-881-SW TORUS |
| RHR-SNUB-(RH-S33D) | R-881-NW TORUS |
| RHR-SNUB-(RH-S34) | R-903-SW |
| RHR-SNUB-(RH-S35) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S36) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S37) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S38) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S39) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S40) | R-903-B RHR HX RM |
| RHR-SNUB-(RH-S41) | R-859-SW QUAD |
| RHR-SNUB-(RH-S42) | R-859-SW QUAD |
| RHR-SNUB-(RH-S43) | R-881-SW TORUS |
| RHR-SNUB-(RH-S44) | R-881-SW QUAD |
| RHR-SNUB-(RH-S45) | R-881-SW QUAD |
| RHR-SNUB-(RH-S48) | R-881-NW QUAD |
| RHR-SNUB-(RH-S49) | R-881-NW QUAD |
| RHR-SNUB-(RH-S51) | R-903-A RHR HX RM |
| RHR-SNUB-(RH-S52) | R-903-A RHR HX RM |

Table 3.6.1

ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

| Snubber | Location |
|---|---|
| RHR-SNUB-(RH-S54) | R-859-NW QUAD |
| RHR-SNUB-(RH-S55) | R-859-NW QUAD |
| RHR-SNUB-(RH-S56) | R-903-A RHR HX RM |
| RHR-SNUB-(RH-S57) | R-903-A RHR HX RM |
| RHR-SNUB-(RH-S59) | R-881-NW TORUS |
| RHR-SNUB-(RH-S65) | R-881-SW QUAD |
| RHR-SNUB-(RH-S66) | R-903-INJ V RM |
| RHR-SNUB-(RH-S76A) | R-881-SW TORUS |
| RHR-SNUB-(RH-S76B) | R-881-SW TORUS |
| RHR-SNUB-(RH-S77) | R-881-SW TORUS |
| RHR-SNUB-(RH-S78A) | R-881-NW TORUS |
| RHR-SNUB-(RH-S78B) | R-881-NW TORUS |
| RHR-SNUB-(RH-S80) | R-881-NW QUAD |
| RHR-SNUB-(RH-S96A) | R-903-NW |
| RHR-SNUB-(RH-S98) | R-881-NW QUAD |
| RWCU-SNUB-(CU-S89) | R-881-SE TORUS |
| SW-SNUB-(SW-H23A) | IS-SWP RM |
| SW-SNUB-(SW-H23D) | IS-SWP RM |
| SW-SNUB-(SW-H23E) | IS-SWP RM |
| SW-SNUB-(SW-H23H) | IS-SWP RM |

Table 3.6.2

ACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)

| Snubber | Location |
|---|---|
| MS-SNUB-(MS-S149B) | R-903-STM TUNNEL |
| MS-SNUB-(MS-S16) | R-881-NW TORUS |
| MS-SNUB-(MS-S9A) | R-881-SW TORUS |
| MS-SNUB-(MS-S9B) | R-881-SW TORUS |
| RCIC-SNUB-(RF-S51C) | R-881-NE TORUS |
| RHR-SNUB-(RH-S58) | R-903-A RHR HX RM |
| SGT-SNUB-(PSSP-40) | R-881-SW TORUS |
| SGT-SNUB-(PSSP-74) | R-881-SW TORUS |
| SW-SNUB-(SW-H23B) | IS-SWP RM |
| SW-SNUB-(SW-H23C) | IS-SWP RM |
| SW-SNUB-(SW-H23F) | IS-SWP RM |
| SW-SNUB-(SW-H23G) | IS-SWP RM |

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)

| Snubber | Location |
|---|---|
| CS-SNUB-(CS-S4) | DW-934 |
| CS-SNUB-(CS-S5) | DW-934 |
| CS-SNUB-(CS-S8) | DW-934 |
| CS-SNUB-(CS-S9) | DW-934 |
| MS-SNUB-(MS-S21) | DW-901 |
| MS-SNUB-(MS-S22) | DW-901 |
| MS-SNUB-(MS-S63) | DW-921 |
| MS-SNUB-(SS-A2) | DW-921 |
| MS-SNUB-(SS-A3) | DW-921 |
| MS-SNUB-(SS-B2) | DW-921 |
| MS-SNUB-(SS-B3) | DW-921 |
| MS-SNUB-(SS-C2) | DW-921 |
| MS-SNUB-(SS-C3) | DW-921 |
| MS-SNUB-(SS-D2) | DW-921 |
| MS-SNUB-(SS-D3) | DW-921 |
| MS-SNUB-(VR-55-23-X) | DW-901 |
| MS-SNUB-(VR-55-26-Z) | DW-901 |
| MS-SNUB-(VR-55-9-Y) | DW-901 |
| MS-SNUB-(VR-55-9-Z) | DW-901 |
| MS-SNUB-(VR-56-12-Y) | DW-901 |
| MS-SNUB-(VR-56-24-X) | DW-901 |
| MS-SNUB-(VR-58-12-Y) | DW-921 |
| MS-SNUB-(VR-59-7-X) | DW-921 |
| MS-SNUB-(VR-59-7-Z) | DW-901 |
| MS-SNUB-(VR-60-7-X) | DW-921 |
| MS-SNUB-(VR-60-7-Z) | DW-901 |
| MS-SNUB-(VR-61-17-X) | DW-901 |
| MS-SNUB-(VR-61-8-X) | DW-901 |
| MS-SNUB-(VR-61-8-Z) | DW-921 |
| MS-SNUB-(VR-62-17-X) | DW-901 |
| MS-SNUB-(VR-62-8-X) | DW-901 |
| MS-SNUB-(VR-62-8-Z) | DW-921 |
| MS-SNUB-(VR-H61D) | DW-888 |
| MS-SNUB-(VR-H62B) | DW-888 |
| MS-SNUB-(VR-H62C) | DW-888 |
| MS-SNUB-(VR-H63B) | DW-888 |
| MS-SNUB-(VR-H63C) | DW-888 |
| MS-SNUB-(VR-H64D) | DW-888 |
| MS-SNUB-(VR-S1) | DW-901 |
| MS-SNUB-(VR-S10) | DW-901 |
| MS-SNUB-(VR-S11) | DW-921 |
| MS-SNUB-(VR-S12) | DW-901 |
| MS-SNUB-(VR-S14) | DW-888 |
| MS-SNUB-(VR-S2) | DW-901 |
| MS-SNUB-(VR-S20) | DW-921 |
| MS-SNUB-(VR-S21) | DW-921 |
| MS-SNUB-(VR-S22) | DW-901 |
| MS-SNUB-(VR-S23A) | DW-901 |
| MS-SNUB-(VR-S23B) | DW-901 |
| MS-SNUB-(VR-S24A) | DW-901 |
| MS-SNUB-(VR-S24B) | DW-901 |
| MS-SNUB-(VR-S25) | DW-901 |

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

| Snubber | Location |
|---|---|
| MS-SNUB-(VR-S26) | DW-888 |
| MS-SNUB-(VR-S27) | DW-901 |
| MS-SNUB-(VR-S3) | DW-888 |
| MS-SNUB-(VR-S30) | DW-921 |
| MS-SNUB-(VR-S31) | DW-921 |
| MS-SNUB-(VR-S32) | DW-888 |
| MS-SNUB-(VR-S4) | DW-901 |
| MS-SNUB-(VR-S40) | DW-921 |
| MS-SNUB-(VR-S41) | DW-921 |
| MS-SNUB-(VR-S42A) | DW-921 |
| MS-SNUB-(VR-S42B) | DW-921 |
| MS-SNUB-(VR-S43) | DW-888 |
| MS-SNUB-(VR-S50A) | DW-921 |
| MS-SNUB-(VR-S50B) | DW-921 |
| MS-SNUB-(VR-S51) | DW-888 |
| MS-SNUB-(VR-S5A) | DW-901 |
| MS-SNUB-(VR-S5B) | DW-901 |
| MS-SNUB-(VR-S6) | DW-901 |
| MS-SNUB-(VR-S60) | DW-921 |
| MS-SNUB-(VR-S61) | DW-921 |
| MS-SNUB-(VR-S62A) | DW-921 |
| MS-SNUB-(VR-S62B) | DW-921 |
| MS-SNUB-(VR-S63) | DW-921 |
| MS-SNUB-(VR-S70A) | DW-901 |
| MS-SNUB-(VR-S70B) | DW-901 |
| MS-SNUB-(VR-S71A) | DW-901 |
| MS-SNUB-(VR-S71B) | DW-901 |
| MS-SNUB-(VR-S72) | DW-901 |
| MS-SNUB-(VR-S73) | DW-901 |
| MS-SNUB-(VR-S74) | DW-901 |
| MS-SNUB-(VR-S7A) | DW-888 |
| MS-SNUB-(VR-S7B) | DW-888 |
| MS-SNUB-(VR-S8) | DW-888 |
| MS-SNUB-(VR-S80) | DW-901 |
| MS-SNUB-(VR-S81) | DW-901 |
| MS-SNUB-(VR-S82) | DW-901 |
| MS-SNUB-(VR-S83A) | DW-901 |
| MS-SNUB-(VR-S83B) | DW-901 |
| MS-SNUB-(VR-S84) | DW-901 |
| MS-SNUB-(VR-S85) | DW-901 |
| MS-SNUB-(VR-S86A) | DW-901 |
| MS-SNUB-(VR-S86B) | DW-901 |
| MS-SNUB-(VR-S87A) | DW-888 |
| MS-SNUB-(VR-S87B) | DW-888 |
| MS-SNUB-(VR-S88) | DW-888 |
| RF-SNUB-(RF-S10) | DW-921 |
| RF-SNUB-(RF-S11) | DW-921 |
| RF-SNUB-(RF-S12) | DW-921 |
| RF-SNUB-(RF-S13) | DW-921 |
| RF-SNUB-(RF-S14) | DW-921 |
| RF-SNUB-(RF-S15) | DW-921 |
| RF-SNUB-(RF-S16) | DW-921 |

Table 3.6.3
INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

| Snubber | Location |
|---|---|
| RF-SNUB-(RF-S17) | DW-921 |
| RF-SNUB-(RF-S18) | DW-921 |
| RF-SNUB-(RF-S19) | DW-921 |
| RF-SNUB-(RF-S8) | DW-921 |
| RF-SNUB-(RF-S9) | DW-921 |
| RHR-SNUB-(RH-S10) | DW-901 |
| RHR-SNUB-(RH-S11) | DW-901 |
| RHR-SNUB-(RH-S13) | DW-921 |
| RHR-SNUB-(RH-S14) | DW-921 |
| RHR-SNUB-(RH-S15) | DW-921 |
| RHR-SNUB-(RH-S16) | DW-901 |
| RHR-SNUB-(RH-S17) | DW-901 |
| RHR-SNUB-(RH-S18) | DW-901 |
| RHR-SNUB-(RH-S19) | DW-901 |
| RHR-SNUB-(RH-S3) | DW-FLG AREA |
| RHR-SNUB-(RH-S4) | DW-FLG AREA |
| RHR-SNUB-(RH-S5) | DW-921 |
| RHR-SNUB-(RH-S6) | DW-921 |
| RHR-SNUB-(RH-S67) | DW-901 |
| RHR-SNUB-(RH-S68) | DW-901 |
| RHR-SNUB-(RH-S69A) | DW-901 |
| RHR-SNUB-(RH-S69B) | DW-901 |
| RHR-SNUB-(RH-S7) | DW-921 |
| RHR-SNUB-(RH-S70) | DW-901 |
| RHR-SNUB-(RH-S71) | DW-901 |
| RHR-SNUB-(RH-S72) | DW-901 |
| RHR-SNUB-(RH-S72A) | DW-901 |
| RHR-SNUB-(RH-S73) | DW-901 |
| RHR-SNUB-(RH-S8A) | DW-901 |
| RHR-SNUB-(RH-S8B) | DW-901 |
| RHR-SNUB-(RH-S9) | DW-901 |
| RR-SNUB-(SS-1A) | DW-888 |
| RR-SNUB-(SS-1B) | DW-888 |
| RR-SNUB-(SS-2A) | DW-888 |
| RR-SNUB-(SS-2B) | DW-888 |
| RR-SNUB-(SS-3A1) | DW-901 |
| RR-SNUB-(SS-3A2) | DW-901 |
| RR-SNUB-(SS-3B1) | DW-901 |
| RR-SNUB-(SS-3B2) | DW-901 |
| RR-SNUB-(SS-4A) | DW-901 |
| RR-SNUB-(SS-4B) | DW-901 |
| RR-SNUB-(SS-5A) | DW-888 |
| RR-SNUB-(SS-5B) | DW-888 |
| RR-SNUB-(SS-8A1) | DW-901 |
| RR-SNUB-(SS-8A2) | DW-901 |
| RWCU-SNUB-(CU-S3A) | DW-921 |
| RWCU-SNUB-(CU-S3B) | DW-921 |

Table 3.6.4

INACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

| Snubber | Location |
|---|---|
| RR-SNUB-(SS-7A1) | DW-901 |
| RR-SNUB-(SS-7A2) | DW-901 |
| RR-SNUB-(SS-7B1) | DW-901 |
| RR-SNUB-(SS-7B2) | DW-901 |

3.7 Containment Systems

Applicability:

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

Objective:

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

Specification:

A. Primary Containment

1. Suppression Pool

At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2. and 3.5.F.5.

a. Minimum water volume - 87,650 $ft^3$

b. Maximum water volume - 91,000 $ft^3$

c. Maximum suppression pool temperature during normal power operation - 95°F.

d. During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in c. above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in c. above within 24 hours.

e. The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in c. above.

4.7 Containment Systems

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool

a. The suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

3.7.A.1 (cont'd.)

f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Containment Integrity

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

4.7.A (cont'd.)

2. Leak Rate Testing

a. Integrated leak rate tests (ILRT's) shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the leakage rate does not exceed the equivalent of 0.635 percent of the primary containment volume per 24 hours at 58 psig.

b. Integrated leak rate tests may be performed at either 58 psig or 29 psig, the leakage rate test period, extending to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed-upon shorter period may be used.

Prior to initial operation, integrated leak rate tests must be performed at 58 and 29 psig (with the 29 psig test being performed prior to the 58 psig test) to establish the allowable leak rate, $L_t$ (in percent of containment volume per 24 hours) at 29 psig as the lesser of the following values:

($L_a$ is 0.635 percent)

$$L_t = 0.635 \frac{L_{tm}}{L_{am}}$$

for $\frac{L_{tm}}{L_{am}} \leq 0.7$

where

$L_{tm}$ = measured ILR at 29 psig

$L_{am}$ = measured ILR at 58 psig, and

$\frac{L_{tm}}{L_{am}} \leq 1.0$

$$L_t = 0.635 \left(\frac{P_t}{P_a}\right)^{1/2}$$

3.7.A (cont'd.)

4.7.A.2.b. (cont'd.)

where

$P_a$ = peak accident pressure, 58 psia

$P_t$ = appropriately measured test pressures (psia)

for $\frac{L_{tm}}{L_{am}} > 0.7$

c. The ILRT's shall be performed at the following minimum frequency:

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to eight months if necessary to coincide with refueling outage.

d. The measured leakage rates, $L_{tm}$ and $L_{am}$, shall be less than 0.75 $L_t$ and 0.75 $L_a$ for the reduced pressure tests and peak pressure test respectively.

e. Except for the initial ILRT, all ILRT's shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If an ILRT has to be terminated due to excessive leakage through identified leakage paths, the leakage through such paths shall be determined by a local leakage test and recorded. After repairs are made another ILRT shall be conducted.

If an ILRT is completed but the acceptance criteria of Specification 4.7.A.2.d is not satisfied and repairs are necessary, the ILRT need not be

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7.A (Cont'd) | 4.7.A.2.e (cont'd) |

repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. <u>Local Leak Rate Tests</u>

1. With the exceptions specified below, local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. Table 3.7.2 specifies testable penetrations with double O-ring seals, Table 3.7.3 specifies testable penetrations with testable bellows, and Table 3.7.4 specifies primary containment testable isolation valves. The test duration of all valves and penetrations shall be of sufficient length to determine repeatable results. The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La.

2. Bolted double-gasket seals (Table 3.7.2) shall be tested after each opening and during each reactor shutdown for refueling, or other convenient intervals but in no case at intervals greater than two years.

3. The main steam isolation valves (MSIV's) shall be tested at a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repairs and retest shall be performed to correct the condition. This is an exemption to Appendix J of 10CFR50.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
| --- | --- |
| 3.7.A (Cont'd) | 4.7.A.2.F (cont'd)<br><br>4. Main steam line and feedwater line expansion bellows as specified in Table 3.7.3 shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50.<br><br>5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50.<br><br>g. Continuous Leak Rate Monitor<br><br>When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.<br><br>h. Drywell Surfaces<br><br>The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage. |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7.A (cont'd.) | 4.7.A (cont'd.) |
| 3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers | 3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers |
| a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc. | a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months. |
| b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker switch shall be secured in the closed position and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity. | b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specifications 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements. |
| 4. Drywell-Pressure Suppression Chamber Vacuum Breakers | 4. Drywell-Pressure Suppression Chamber Vacuum Breakers |
| a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below. | a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days. |
| b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met. | b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service. |

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| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7.A.4 (cont'd.) | 4.7.A.4 (cont'd.) |
| c. The total leakage between the dry-well and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice. | c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation of the position indication switch. The differential pressure setpoint shall be verified. |
| d. If specifications 3.7.A.4.a, b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours. | d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met. |
| 5. Oxygen Concentration | 5. Oxygen Concentration |
| a. After completion of the startup test program and demonstration of plant electrical output, the primay containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b. | a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly. |
| b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown. | b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect. |
| c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons. | |
| d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. | |
| e. The specifications of 3.7.A.5.a thru d are not applicable during a 48 hour continuous period between the dates of March 22, 1982 and March 25, 1982. | |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7. (cont'd.) | 4.7 (cont'd.) |
| B. Standby Gas Treatment System | B. Standby Gas Treatment System |
| 1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required. | 1. At least once per operating cycle the following conditions shall be demonstrated.<br><br>a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.<br><br>b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H. |
| 2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal. | 2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system. |
| b. The results of laboratory carbon sample analysis shall show $\geq$99% radioactive methyl iodide removal at a velocity within 20 percent of actual system design, $\geq$1.75 mg/m$^3$ inlet methyl iodide concentration, $\geq$70% R.H. and $\leq$30$^\circ$F. | b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing. |
| c. Fans shall be shown to operate within $\pm$10% design flow. | c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.<br><br>d. Each circuit shall be operated with the heaters on at least 10 hours every month.<br><br>e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a. |
| 3. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation or fuel handling is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit shall be operable. | 3. System drains where present shall be inspected quarterly for adequte water level in loop-seals. |

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 1)
PRIMARY CONTAINMENT ISOLATION VALVES

| Valve & Steam | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (Sec) (1) | Normal Position (2) | Action On Initiating Signal (3) |
|---|---|---|---|---|---|
| Main Steam Isolation Valves | | | | | |
| MS-AO-80- A,B,C, & D | 4 | | $3 < T < 5$ | 0 | GC |
| MS-AO-86- A,B,C, & D | | 4 | $3 < T < 5$ | 0 | GC |
| Drywell Floor Drain Iso. Valves RW-AO-82, RW-AC-83 | 1 | 1 | 15 | 0 | GC |
| Drywell Equipment Drain Iso. Valves RW-AO-94, RW-AO-95 | 1 | 1 | 15 | 0 | GC |
| Main Steam Line Drain Valves MS-MO-74, MS-MO-77 | 1 | 1 | 30 | C | SC |
| Reactor Water Sample Valves RRV-740AV, RRV-741AV | 1 | 1 | 15 | 0 | GC |
| Reactor Water Cleanup System Iso. Valves RWCU-MO-15, RWCU-MO-18 | 1 | 1 | 60 | 0 | GC |
| RHR Reactor Head Spray Iso. Valves RHR-MO-32, RHR-MO-33 | 1 | 1 | 60 | C | SC |
| RHR Suction Cooling Iso. Valve RHR-MO-17, RHR-MO-18 | 1 | 1 | 40 | C | SC |
| RHR Discharge to Radwaste Iso. Valves RHR-MO-57, RHR-MO-67 | 1 | 1 | 20 | C | SC |
| Suppression Chamber Purge & Vent PC-245AV, PC-230MV | | 2 | 15 | C | SC |
| Suppression Chamber $N_2$ Supply PC-237AV, PC-233MV | | 2 | 15 | C | SC |

TABLE 3.7.4
PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

| PEN. NO. | VALVE NUMBERS | TEST MEDIA |
|---|---|---|
| X-7A | MS-AO-80A and MS-AO-86A, Main Steam Isolation Valves | Air |
| X-7B | MS-AO-80B and MS-AO-86B, Main Steam Isolation Valves | Air |
| X-7C | MS-AO-80C and MS-AO-86C, Main Steam Isolation Valves | Air |
| X-7D | MS-AO-80D and MS-AO-86D, Main Steam Isolation valves | Air |
| X-8 | MS-MO-74 and MS-MO-77, Main Steam Line Drain | Air |
| X-9A | RF-15CV and RF-16CV, Feedwater Check Valves | Air |
| X-9A | RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater | Air |
| X-9B | RF-13CV and RF-14CV, Feedwater Check Valves | Air |
| X-9B | HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater | Air |
| X-10 | RCIC-MO-15 and RCIC-MO-16, RCIC Steam Line | Air |
| X-11 | HPCI-MO-15 and HPCI-MO-16, RPCI Steam Line | Air |
| X-12 | RHR-MO-17 and RHR-MO-18, RHR Suction Cooling | Air |
| X-13A | RHR-MO-25A and RHR-MO-27A, RHR Supply to RPV | Air |
| X-13B | RHR-MO-25B and RHR-MO-27B, RHR Supply to RPV | Air |
| X-14 | RWCU-MO-15 and RWCU-MO-18, Inlet to RWCU System | Air |
| X-16A | CS-MO-11A and CS-MO-12A, Core Spray to RPV | Air |
| X-16B | CS-MO-11B and CS-MO-12B, Core Spray to RPV | Air |
| X-17 | RHR-MO-32 and RHR-MO-33, RPV Head Spray | Air |
| X-18 | RW-732AV and RW-733AV, Drywell Equipment Sump Discharge | Air |
| X-19 | RW-765AV and RW-766AV, Drywell Floor Drain Sump Discharge | Air |
| X-25 | PC-232MV and PC-238AV, Purge and Vent Supply to Drywell | Air |
| X-25 | ACAD-1305MV and ACAD-1306MV, Supply to Drywell | Air |
| X-26 | PC-231MV and PC-246AV, Purge and Vent Exhaust from Drywell | Air |
| X-26 | ACAD-1310MV, Bleed from Drywell | Air |

3.7.A & 4.7.A BASES

Primary Containment

The integrity of t e primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 58 psig which is below the maximum of 62 psig. Maximum water volume of 91,000 $ft^3$ results in a downcomer submergence of 5' and the minimum volume of 87,650 $ft^3$ results in a submergence approximately 12 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in bases 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The maximum suppression pool temperature of 95°F is based on not exceeding the 200°F Mark I temperature limit as contained in NUREG-0661. This 95°F limit also prevents exceeding LOCA considerations, or ECCS pump NPSH requirements. The basis for these limits are contained in NEDC-24360-P.

Inerting

Safety Guide 7 assumptions for Metal-Water reaction result in hydrogen concentration in excess of the Safety Guide 7 flammability limit. By keeping the oxygen concentration less than 4% by volume the requirements of Safety Guide 7 are satisfied.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during periods when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

### Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the standby gas treatment system is not required.

## 4.7.B & 4.7.C BASES

### Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with ANSI N510-1980. The test cannisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

## 3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

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

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of $1.1 \times 10^{-7}$ that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

    a. Observe flow cessation and any leakage rate.

    b. Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed and the leakage rate is acceptable.

3.7.E Bases

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed as described in the licensee's letter of October 4, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of drywell-suppression chamber differential pressure of 1.0 psid and a suppression chamber water level corresponding to a downcomer submergence range of three to four feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

## 3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical from a Cold Shutdown Condition unless all of the following conditions are satisfied:

1. Both off-site sources (345 KV and 69 KV) and the startup transformer and emergency transformer are available and capable of automatically supplying power to the 4160 Volt emergency buses 1F and 1G.

2. Both diesel generators shall be operable and there shall be a minimum of 45,000 gal. of diesel fuel in the fuel oil storage tanks.

3. The 4160V critical buses 1F and 1G and the 480V critical buses 1F and 1G are energized.

    a. The loss of voltage relays and their auxiliary relays are operable.

    b. The undervoltage relays and their auxiliary relays are operable.

4. The four unit 125V/250V batteries and their chargers shall be operable.

5. The power monitoring system for the inservice RPS MG set or alternate source shall be operable.

## 4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment

1. Emergency Buses Undervoltage Relays

    a. Loss of voltage relays

    Once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators.

    b. Undervoltage relays

    Once every 18 months, low voltage on emergency buses is simulated to demonstrate disconnection of the emergency buses from the offsite power source. The undervoltage relays shall be calibrated once every 18 months.

2. Diesel Generators

    a. Each diesel-generator shall be started manually and loaded to not less than 35% of rated load for no less than 2 hours once each month to demonstrate operational readiness.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

3.9.A

4.9.A (cont'd.)

cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of the voltage of each cell to nearest 0.1 Volt, specific gravity of each cell, and temperature of every sixth cell. These measurements shall be logged.

c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1.

1. From and after the date incoming power is not available from a startup or emergency transformer, continued reactor operation is permissible under this condition for seven days. At the end of this period, provided the second source of incoming power has not been made immediately available, the NRC must be notified of the event and the plan to restore this second source. During this period, the two diesel generators and associated critical buses must be demonstrated to be operable.

2. From and after the date that incoming power is not available from both start-up and emergency transformers, continued operation is permissible, provided the two diesel generators and associated critical buses are demonstrated to be

4. Power Monitoring System for RPS System

The above specified RPS power monitoring system instrumentation shall be determined operable:

a. At least once per operating cycle by demonstrating the operability of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a channel calibration including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage $\leq$ 132 VAC, with time delay $\leq$ 2 sec.

   2. Under-voltage $\geq$ 108 VAC, with time delay $\leq$ 2 sec.

   3. Under-frequency $\geq$ 57 Hz. with time delay $\leq$ 2 sec.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
| --- | --- |
| 3.9.B.5 (cont'd.)<br><br>From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding ten days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specifications 3.5.A.5 and 3.5.F are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed components to an operable state.<br><br>6. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.<br><br>7. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service. | 4.9.B |

## 4.9 BASES

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel generator will be manually started, synchronized and connected to the bus and load picked up. The diesel generator should be loaded to at least 35% of rated load to prevent fouling of the engine. It is expected that the diesel generator will be run for at least two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required.

Each diesel generator has two air compressors and two air receivers for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the diesel generator, each receiver in each set of receivers will be drawn down below the point at which the corresponding compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

The diesel generator fuel consumption rate at full load is approximately 275 gallons per hour. Thus, the monthly load test of the diesel generators will test the operation and the ability of the fuel oil transfer pumps to refill the day tank and will check the operation of these pumps from the emergency source.

The test of the diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e, it will check diesel generator starting and closure of diesel generator breaker and sequencing of load on the diesel generator. The diesel generator will be started by simulation of a loss-of-coolant accident. In addition, an undervoltage condition will be imposed to simulate a loss of off-site power.

Periodic tests between refueling outages verify the ability of the diesel generator to run at full load and the core and containment cooling pumps to deliver full flow. Periodic testing of the various components, plus a functional test once-a-cycle, is sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure. In addition, the checks described also provide adequate indication that the batteries have the specified ampere-hour capability.

The diesel fuel oil quality must be checked to ensure proper operation of the diesel generators. Water content should be minimized because water in the fuel could contribute to excessive damage to the diesel engine.

When it is determined that some auxiliary electrical equipment is out of service, the increased surveillance required in Section 4.5.F is deemed adequate to provide assurance that the remaining equipment will be operable.

The Reactor Protection System (RPS) is equipped with a seismically qualified, Class 1E power monitoring system. This system consists of eight Electrical Protection Assemblies (EPA) which isolate the power sources from the RPS if the input voltage and frequency are not within limits specified for safe system operation. Isolation of RPS power causes that RPS division to fail safe.

LIMITING CONDITIONS FOR OPERATION

## 3.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the operating status of the main control room ventilation system, the reactor building closed cooling water system and the service water system.

Objective:

To assure the availability of the main control room ventilation system, the reactor building closed cooling water system and the service water system upon the conditions for which the capability is an essential response to station abnormalities.

A. Main Control Room Ventilation

1. Except as specified in Specification 3.12.A.3 below, the control room air treatment system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.

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

b. The results of laboratory carbon sample analysis shall show $\geq$ 99% radioactive methyl iodide removal at a velocity within 20% of system design, 1.75 $mg/m^3$ inlet iodide concentration, $\geq$ 95% R.H. and $\leq$30°F.

c. Fans shall be shown to operate within $\pm$ 10% design flow.

SURVEILLANCE REQUIREMENTS

## 4.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water system and the service water system which are required by the corresponding Limiting Conditions for Operation.

Objective:

To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.

A. Main Control Room Ventilation

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal absorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.

2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

## 3.12 BASES

### A. Main Control Room Ventilation System

The control room ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is not immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours, or refueling operations are terminated.

### B. Reactor Building Closed Cooling Water System

The reactor building closed cooling water system has two pumps and one heat exchanger in each of two loops. Each loop is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either loop.

The system has additional flexibility provided by the capability of interconnection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of LPCI or the core spray systems.

### C. Service Water System

The service water system consists of four vertical service water pumps located in the intake structure, and associated strainers, piping, valving and instrumentation. The pumps discharge to a common header from which independent piping supplies two Seismic Class I cooling water loops and one turbine building loop. Automatic valving is provided to shutoff all supply to the turbine building loop on drop in header pressure thus assuring supply to the Seismic Class I loops each of which feeds one diesel generator, two RHR service water booster pumps, one control room basement fan coil unit and one RBCCW heat exchanger. Valves are included in the common discharge header to permit the Seismic Class I service water system to be operated as two independent loops. The heat exchangers are valved such that they can be individually backwashed without interrupting system operation.

During normal operation two or three pumps will be required. Three pumps are used for a normal shutdown.

The loss of all a-c power will trip all operating service water pumps. The automatic emergency diesel generator start system and emergency equipment starting sequence will then start one selected service water pump in 30-40 seconds. In the meantime, the drop in service water header pressure will close the turbine building cooling water isolation valve guaranteeing supply to the reactor building, the control room basement, and the diesel generators from the one service water pump.

Due to the redundance of pumps and the requirement of only one to meet the accident requirements, the 30 day repair time is justified.

D. Battery Room Ventilation

The temperature rise and hydrogen buildup in the battery rooms without adequate ventilation is such that continuous safe operation of equipment in these rooms cannot be assured.

4.12 BASES

A. Main Control Room Ventilation System

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant should be performed in accordance with ANSI N510-1980.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The test cannisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the absorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

## LIMITING CONDITIONS FOR OPERATION

### 3.14 FIRE DETECTION SYSTEM

APPLICABILITY

Applies to the operational status of the Fire Detection System.

OBJECTIVE

To assure continuous automatic surveillance throughout the Main Plant.

SPECIFICATIONS

A. The Fire Detection System instumentation for each fire detection zone shown in Table 3.14 shall be operable.

B. With one or more of the fire detection instrument(s) shown in Table 3.14 inoperable:

1. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.7.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

### 3.15 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY

Applies to the availability of water for fire fighting purposes.

OBJECTIVE

To assure a continuous operable water supply for fire fighting systems from 2 fire pumps.

## SURVEILLANCE REQUIREMENTS

### 4.14 FIRE DETECTION SYSTEM

APPLICABILITY

Applies to the operational status of the Fire Detection System.

SPECIFICATIONS

A. Each detector on Table 3.14 shall be demonstrated operable every 6 months by performance of a channel functional test.

B. The NFPA Code 72.D Class B 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.

### 4.15 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY

Applies to the availability of water for fire fighting purposes.

## LIMITING CONDITIONS FOR OPERATION

3.15 (cont'd)

### SPECIFICATIONS

A. The fire suppression water system shall be OPERABLE with:

1. Two fire pumps, each with a capacity of at least 2000 gpm, with their discharge aligned to the fire suppression header.

2. An OPERABLE flow path capable of taking suction from either of two 500,000 gallon water storage tanks or the Missouri River and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant valves and the front valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

B. If the requirement of 3.15.A cannot be met, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.7.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

C. With the fire suppression system inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.7.2;

   a) By telephone within 24 hours, and

   b) In writing no later than the first working day following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

## SURVEILLANCE REQUIREMENTS

4.15 (cont'd)

### SPECIFICATIONS

A. The Fire Suppression Water Supply System shall be demonstrated operable:

1. At least once per 31 days by starting each pump on a staggered start-up basis and operating it for:

   a) A minimum of 15 minutes for a diesel engine-driven fire pump, and

   b) A minimum of 7 minutes for an electrical motor-driven fire pump.

2. 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.

3. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

4. 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:

   a) Verifying that each automatic valve in the flow path actuates to its correct position on a test signal,

   b) Verifying that each pump developes at least 2000 gpm with at least 110 psi,

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

4.15 (cont'd)

c) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle or full travel, and

d) Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure $\geq$ 65 psig.

5. 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.

6. The fire pump diesel engine shall be demonstrated OPERABLE:

   a) At least once per 31 days by verifying;

      1) The fuel storage tank contains at least 150* gallons of fuel, and

      2) The diesel starts from ambient conditions and operates for at least 15 minutes.

   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-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 for viscosity water content and sediment.

   c) At least once per 18 months by:

      1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

*This number shall become 250 gallons when the clean water fire protection system becomes operable.

| INSTRUMENT LOCATION | | INSTRUMENT ID NO. |
|---|---|---|
| 2 | <u>Control Room</u> | FP-SD-17-1<br>FP-SD-17-2<br>FP-SD-17-3 |
| 3 | <u>Cable Spreading Room</u> | FP-SD-16-1<br>FP-SD-16-2<br>FP-SD-16-3<br>FP-SD-16-4<br>FP-SD-16-5<br>FP-SD-16-6 |
| | <u>Cable Expansion Room</u> | FP-SD-16-7<br>FP-SD-16-8 |
| 4 | <u>Switchgear Rooms</u> | |
| | DC Switchgear Rooms | FP-SD-15-2<br>FP-SD-15-3 |
| | Critical Switchgear Room | FP-SD-22-1<br>FP-SD-22-2 |
| 5 | <u>Station Battery Rooms</u> | FP-SD-15-1<br>FP-SD-15-4<br>FP-SD-15-1A<br>FP-SD-15-4A |
| 6 | <u>Diesel Generator Rooms</u> | FP-SD-10-1<br>FP-SD-10-2<br>FP-SD-10-3<br>FP-SD-10-4<br>CO2-SD-DG-1A<br>CO2-SD-DG-1B<br>CO2-SD-DG-1C<br>CO2-SD-DG-1D<br>CO2-SD-DG-2A<br>CO2-SD-DG-2B<br>CO2-SD-DG-2C<br>CO2-SD-DG-2D |
| 7 | <u>Diesel Fuel Storage Rooms</u> | CO2-TD-DG-1A<br>CO2-TD-DG-1B |
| 8 | <u>Safety Related Equipment not in Reactor Building</u> | |
| | RHR Service Water Booster Pumps | FP-SD-14-3 |
| | Emergency Condensate Storage Tanks | FP-SD-14-1 |
| | Service Water Pumps | FP-FD-32-1<br>FP-FD-32-2 |
| 9 | <u>Auxiliary Relay Room & Reactor Protection System Rooms</u> | |
| | Auxiliary Relay Room | FP-SD-15-9 |
| | Reactor Protection System Room 1A | FP-SD-15-7 |
| | Reactor Protection System Room 1B | FP-SD-15-8 |

## 6.0 ADMINISTRATIVE CONTROLS

## 6.1 ORGANIZATION

### 6.1.1 Responsibility

The Station Superintendent shall have the over-all fulltime onsite responsibility for the safe operation of the Cooper Nuclear Station. During periods when the Station Superintendent is unavailable, he may delegate his responsibility to the Assistant to Station Superintendent or, in his absence, to one of the Department Supervisors.

### 6.1.2 Offsite

The portion of the Nebraska Public Power District management which relates to the operation of this station is shown in Figure 6.1.1.

### 6.1.3 Plant Staff - Shift Complement

The organization for conduct of operation of the station is shown in Fig. 6.1.2. The shift complement at the station shall at all times meet the following requirements. Note: Higher grade licensed operators may take the place of lower grade licensed or unlicensed operators.

A. A licensed senior reactor operator (SRO) shall be present at the station at all times when there is any fuel in the reactor.

B. A licensed reactor operator shall be in the control room at all times when there is any fuel in the reactor.

C. Two licensed reactor operators shall be in the control room during all startup, shutdown and other periods involving significant planned control rod manipulations. A licensed SRO shall either be in the Control Room or immediately available to the Control Room during such periods.

D. A licensed senior reactor operator (SRO) with no other concurrent duties shall be directly in charge of any refueling operation, or alteration of the reactor core.

A licensed reactor operator (RO) with no other concurrent duties shall be directly in charge of operations involving the handling of irradiated fuel other than refueling or reactor core alteration operations.

E. An individual who has been trained and qualified in health physics techniques shall be on site at all times that fuel is on site.

F. Minimum crew size during reactor operation shall consist of three licensed reactor operators (one of whom shall be licensed SRO) and three unlicensed operators. Minimum crew size during reactor cold shutdown conditions shall consist of two licensed reactor operators (one of whom shall be licensed SRO) and one unlicensed operator.

In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury a qualified replacement shall be designated to report on-site within two hours.

G. A Fire Brigade of at least 5 members shall be maintained at all times. This excludes the 3 members of the minimum shift crew necessary for safe shutdowns, and other personnel required for other essential functions during a fire emergency. Three fire Brigade members shall be from the Operations Department and 2 support members may be from other departments inclusive of Security personnel.

Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

6.1.4 Plant Staff - Qualifications

The minimum qualifications, training, replacement training, and retraining of plant personnel at the time of fuel loading or appointment to the active position shall meet the requirements as described in the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants". The Assistant to Station Superintendent qualifications shall comply with Section 4.2 of ANSI-N18.1-1971. The Chemistry and Health Physics Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, Sept. 1975; personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

A. A training program for the fire brigade will be maintained under the direction of the plant training coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code 1976, except for Fire Brigade training sessions which shall be held at least quarterly.

The training program requirements will be provided by a qualified fire protection engineer.

## 6.2 REVIEW AND AUDIT

6.2.1 The organization and duties of committees for the review and audit of station operation shall be as outlined below:

A. Station Operations Review Committee (SORC)

1. Membership:

   a. Chairman: Station Superintendent or Assistant to Station Superintendent

   b. Engineering Supervisor

   c. Operations Supervisor

   d. Chemistry and Health Physics Supervisor

   e. Maintenance Supervisor

   f. Quality Assurance Supervisor - non-voting member.

   Alternate members shall be appointed in writing by the Station Superintendent to serve on a temporary basis; however, no more than one alternate shall serve on the Committee at any one time.

2. Meeting Frequency: Monthly, and as required on call of the Chairman.

3. Quorum: Station Superintendent or Assistant to Station Superintendent plus two other members including alternates.

4. Responsibilities:

   a. Review all proposed normal, abnormal, maintenance and emergency operating procedures specified in 6.3.1, 6.3.2, 6.3.3, and 6.3.4 and proposed changes thereto: and any other proposed procedures or changes thereto determined by any member to effect nuclear safety.

   b. Review all proposed tests and experiments and their results, which involve nuclear hazards not previously reviewed for conformance with technical specifications. Submit tests which may constitute an unreviewed safety question to the NPPD Safety Review and Audit Board for review.

   c. Review proposed changes to Technical Specifications.

   d. Review proposed changes or modifications to station systems or equipment as discussed in the SAR or which involves an unreviewed safety question as defined in 10CFR50.59(c). Submit changes to equipment or systems having safety significance to the NPPD Safety Review and Audit Board for review.

   e. Review station operation to detect potential nuclear safety hazards.

f. Investigate all reported instances of violations of Technical Specifications, including reporting evaluation and recommendations to prevent recurrence, to the Division Manager of Power Operations and to the Chairman of the NPPD Safety Review and Audit Board.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review and Audit Board.

h. Review all events which are required by Technical Specifications to be reported to the NRC in writing within 24 hours.

i. Review drills on emergency procedures (including plant evacuation) and adequacy of communication with off site groups.

j. Periodically review procedures required by Specifications 6.3.1, 6.3.2, 6.3.3, and 6.3.4 as set forth in administrative procedures.

5. Authority

a. The Station Operations Review Committe shall be advisory.

b. The Station Operations Review Committee shall recommend to the Station Superintendent approval or disapproval of proposals under items 4, a through e and j above. In case of disagreement between the recommendations of the Station Operations Review Committee and the Station Superintendent, the course determined by the Station Superintendent to be the more conservative will be followed. A written summary of the disagreement will be sent to the Division Manager of Power Operations and to the NPPD Safety Review and Audit Board.

c. The Station Operations Review Committee shall report to the Chairman of the NPPD Safety Review and Audit Board on all reviews and investigations conducted under items 4.f, 4.g, 4.h, and 4.i.

d. The Station Operations Review Committee shall make tentative determinations regarding whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review and approval by the NPPD Safety Review and Audit Board.

6. Records:

Minutes shall be kept for all meetings of the Station Operations Review Committee and shall include identification of all documentary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Director of Power Supply within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. <u>NPPD Safety Review and Audit Board (SRAB)</u>

Function: The Board shall function to provide independent review and audit of designated activities.

6.2 (cont'd)

1. Membership:

   a. Senior Division Manager of Power Operations (chairman)

   b. Division Manager of Licensing and Quality Assurance (alternate Chairman)

   c. Division Manager of Power Projects

   d. Division Manager of Power Operations

   e. Division Manager of Environmental Affairs

   f. Consultants (as required)

   The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized.

   Alternate members shall be appointed in writing by the Board Chairman to serve on a temporary basis; however, no more than two alternates shall serve on the Board at any one time.

2. Meeting frequency: Semiannually, and as required on call of the Chairman.

3. Quorum: Chairman or Vice Chairman, plus three members including alternates. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Review: The following subjects shall be reported to and reviewed by the NPPD Safety Review and Audit Board.

   a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

   b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or this Operating License.

e. Violations of applicable 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 plant equipment that affect nuclear safety.

g. All events which are required by Technical Specifications to be reported to the NRC in writing within 24 hours.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Minutes of meetings of the Station Operations Review Committee.

j. Disagreement between the recommendations of the Station Operations Review Committee and the Station Superintendent.

k. Review of events covered under e,f,g, and h above include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority: The NPPD Safety Review and Audit Board shall be advisory to the General Manager and shall have authority to:

   a. Approve proposed changes to the operating license including Technical Specifications and Safety Analysis Report for submission to the NRC.

b. Approve safety related changes or modifications to station systems and equipment, provided such changes of modifications do not involve an unreviewed safety question by the NRC and do not require changes in the Operating License.

6. Records:

Minutes shall be recorded for all meetings of the NPPD Safety Review and Audit Board and shall identify all documentary material reviewed. Copies of the minutes shall be forwarded to the General Manager and the Station Superintendent, and such others as the Chairman may designate within one month of the meeting.

7. Procedures:

Written administrative procedures for Board operation shall be prepared and maintained that contain:

a. Subjects within the purview of the group.

b. Responsibility and authority of the group including responsibility to identify problems; and to recommend solutions, and authority to verify implementaton of approved actions.

c. Mechanisms for convening meetings.

d. Provisions for any use of subgroups.

e. Authority to obtain access to the nuclear power plant operating record files and operating personnel to perform the audit function.

f. Requirements for distribution of reports and minutes prepared by the group to others in the NPPD organization.

g. Identification of the management position to which the Board reports.

h. Provisions for assuring that the Committee is kept informed on a timely basis of matters within its purview.

i. Provision for a formal approval of the minutes.

8. Audits:

Audits of selected aspects of plant operation shall be performed under the cognizance of SRAB with a frequency commensurate with their safety significance. Audits performed by the Quality Assurance Department which meet this specification shall be considered to meet the SRAB audit requirements if the audit results are reviewed by SRAB. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. These audits shall encompass:

6.2 (cont'd)

a. Verification of compliance with internal rules, procedures (for example: normal, off-normal, emergency, operating, maintenance, surveillance, test, and radiation control procedures) and applicable license conditions at least once per 24 months.

b. The training, qualification, and performance of the operating staff at least once per 24 months.

c. The Emergency Plan and implementing procedures at least once per 12 months.

d. The Security Plan and implementing procedures at least once per 12 months.

e. The facility fire protection and its implementing procedures at least once per 24 months.

f. A fire protection and loss prevention inspection will be performed utilizing either qualified off-site licensee personnel or an outside fire protection consultant at least once per 12 months.

g. An inspection and audit by an outside qualified fire protection consultant shall be performed at least once per 36 months.

## 6.3 PROCEDURES AND PROGRAMS

### 6.3.1 Introduction

Station personnel shall be provided detailed written procedures to be used for operation and maintenance of system components and systems that could have an effect on nuclear safety.

### 6.3.2 Procedures

Written procedures and instructions including applicable check off lists shall be provided and adhered to for the following:

A. Normal startup, operation, shutdown and fuel handling operations of the station including all systems and components involving nuclear safety.

B. Actions to be taken to correct specific and forseen potential or actual malfunctions of safety related systems or components including responses to alarms, primary system leaks and abnormal reactivity changes.

C. Emergency conditions involving possible or actual releases of radioactive materials.

D. Implementing procedures of the Security Plan and the Emergency Plan.

E. Implementing procedures for the fire protection program.

F. Administrative procedures for shift overtime.

### 6.3.3 Maintenance and Test Procedures

The following maintenance and test procedures will be provided to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

A. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

B. Routine testing of standby and redundant equipment.

C. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

D. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

E. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

### 6.3.4 Radiation Control Procedures

Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR 20.

6.3 (cont'd)

A. High Radiation Areas

In lieu of the "control device" or "alarm signal" required by Paragraph 20.203 (c) (2) of 10 CFR 20 each High Radiation Area (100 mrem/hr or greater) shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring notification and permission of the shift supervisor. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

6.3.5 Temporary Changes

Temporary changes to procedures which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the operating staff holding SRO licenses. Such changes shall be documented and subsequently reviewed by the Station Superintendent within one month.

6.3.6 Exercise of Procedures

Drills of the Emergency Plan procedures shall be conducted annually, including a check of communications with offsite support groups. Drills on the procedures specified in 6.3.2.A, B, and C above shall be conducted as part of the retraining program.

6.3.7 Programs

The following programs shall be established:

A. Systems Integrity Monitoring Program

A program shall be established to reduce leakage from systems outside the primary containment that would or could contain highly radioactive fluids during a serious accident to as low as practical levels. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

B. Iodine Monitoring Program

A program shall be established to ensure that capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

C. Environmental Qualification Program

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff

Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-46 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

## 6.4 RECORD RETENTION

### 6.4.1 5 Year Retention

Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years unless a longer period is required by applicable regulations.

A. Records of normal station operation, including power levels and periods of operation at each power level.

B. Records of periodic checks, inspection and/or calibrations performed to verify that Surveillance Requirements are being met.

C. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

D. Records of reportable occurrences as specified in 6.5.2.

E. Record of changes to plant procedures.

F. Records of special tests and experiments.

G. Records of wind speed and direction.

### 6.4.2 Life Retention

Records and logs relating to the following items shall be kept for the life of the plant.

A. Records of changes made to the station as described in the Safety Analysis Report and amendments and reflected in updated, corrected and as-built drawings and records.

B. Records of new and spent fuel inventory and assembly histories.

C. Records of station radiation and contamination surveys.

D. Records of off-site environmental monitoring surveys.

E. Records of radiation exposure for all station personnel, including all contractors and visitors to the station in accordance with 10 CFR 20.

F. Records of radioactivity in liquid and gaseous wastes released to the environment.

G. Design Fatigue Usage Evaluation

1. Monitoring, recording, and evaluation will be met for various portions of the reactor coolant pressure boundary (RCPB) for which detailed fatigue

usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The locations to be monitored shall be:

a. The feedwater nozzles

b. The shell at or near the waterline

c. The flange studs

2. Monitoring, Recording, Evaluating, and Reporting

a. Operational transients that occur during plant operations will, at least annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report Code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specification quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicated number of design condition events by twenty-five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

H. Records of individual plant staff members showing qualifications, training and retraining.

I. Records for Environmental Qualification which are covered under the provisions of Specification 6.3.

J. Records of the service lives of all hydraulic and mechanical snubbers, listed on Tables 3.6.1, 3.6.2, 3.6.3, 3.6.4 including the date at which the service life commences and associated installation and maintenance records.

6.4.3 2 Year Retention

Records and logs relating to the following items shall be kept for two years.

A. The test results, in units of microcuries, for leak tests of sources performed pursuant to Specification 3.8.A.

B. Records of annual physical inventories verifying accountability of the sources on record.

1. See paragraph N-415.2, ASME Section III, 1965 Edition.

2. The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition.)

## 6.5 STATION REPORTING REQUIREMENTS

### 6.5.1 Routine Reports

A. Introduction - In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the individual(s) designated in the current revision of Reg. Guide 10.1 unless otherwise noted.

B. Startup Report

1. A summary report of plant startup and power escalation testing shall be submitted following:

   a. Receipt of an operating license.

   b. Amendment to the license involving a planned increase in power level.

   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.

   d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

   The 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.

2. 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 all three events are not completed, supplementary reports shall be submitted every three months.

C. Annual Reports

Routine reports covering the subjects noted in 6.5.1.C.1 6.5.1.C.2, and 6.5.1.C.3 for the previous calendar year shall be submitted prior to March 1 of each year.

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling 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 shall be assigned to specific major work functions.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

6.5.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1/ This tabulation supplements the requirements of §20.407 of 10CFR Part 20.

A. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the appropriate Regional Office, no later than the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 6.5.2.A.5, 6.5.2.A.6 or 6.5.2.B.1 below.

2. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.5.2.B.2 below.

3. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

4. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of any unplanned criticality.

5. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

6. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

   Note: For items 6.5.2.A.5 and 6.5.2.A.6 reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.5.2.B.2 and 6.5.2.B.3 below.

7. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

8. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

9. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

B. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

2. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.5.2.B.1 and 6.5.2.B.2 need not be reported except where test results themselves reveal a degraded mode as described above.

3. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

4. Abnormal degradation of systems other than those specified in item 6.5.2.A.3 above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.5.3. <u>Unique Reporting Requirements</u>

Reports shall be submitted to the Director, Nuclear Reactor Regulation, USNRC, Washington, DC 20555, as follows:

A. Reports on the following areas shall be submitted as noted:

None.