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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-295

ZION STATION UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.62
License No. DPR-39

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated April 22, 1980, May 30, 1980, and October 3, 1980, 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 Commmssion;

   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.

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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-39 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1981

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-304

ZION STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
License No. DPR-48

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated April 22, 1980, May 30, 1980, and October 3, 1980, 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 Commmssion;

   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-48 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 59 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 7, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-39

AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-48

DOCKET NOS. 50-295 AND 50-304

Revise Appendix A as follows:

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ATTACHMENT TO LICENSE AMENDMENTS

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Amendment Nos. 62 & 59

G. Quadrant Power Tilt Ratio

The quadrant power tilt ratio is defined as the ratio of the maximum upper excore detector current to the average of the upper excore detector currents or the ratio of the maximum lower excore detector current to the average of the lower excore detector currents whichever is greater.

H. Rated Thermal Power

A steady-state reactor core output of 3250 $MW_t$ per unit.

I. Reactor Pressure

The pressure in the steam space of a pressurizer.

J. Refueling Outage

When Refueling Outage is used to designate a surveillance interval per unit, the surveillance will be performed during the refueling outage or up to six months before the refueling outage. When a refueling outage occurs within 8 months of the previous refueling outage for a unit, the surveillance testing need not be performed. The maximum interval between surveillance tests is 20 months per unit.

K. Operable

Properly installed in the system and capable of performing the intended functions in the intended manner as verified by testing and tested at the frequency required by the Technical Specifications.

L. Operating

Performing the intended functions in the intended manner.

M. Operating Cycle

The interval between the end of one major refueling outage and the end of the next subsequent major refueling outage per unit.

N. Surveillance Interval

Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval; and

b. A total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified interval.

| SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING |
| --- | --- |
| | 7. Low reactor coolant pump motor frequency: $\geq$ 57.0 cps.<br><br>8. Undervoltage to reactor coolant pump motors: $\geq$ 70% of normal.<br><br>C. Other reactor trips<br><br>1. High pressurizer water level: $\leq$ 92% of span.<br><br>2. Low-low steam generator water level: $\geq$ 10% of narrow range instrument.<br><br>3. Steam feedwater flow mismatch: $\leq$ 60% of nominal 100% steam flow rate in coincidence with low steam generator water level - $\geq$ 10% of narrow range instrument span.<br><br>4. Safety Injection - Trip settings for safety injection are detailed in Section 3.4.<br><br>5. Turbine Trip<br><br>6. Power range, positive high neutron flux rate $\leq$ 15% of rated flux in 5 sec.<br><br>7. Power range, negative high neutron flux rate $\leq \lvert -15\% \rvert$ of the rated flux in 5 sec.<br><br>8. Manual reactor trip. |

The curves are based on the following nuclear hot channel factors: (2)

$$F^N_{\Delta H} = 1.55 \left[1 + 0.2 (1-P)\right]$$

where P is the fraction of rated power, and on a DNB analysis as described in Reference 3. The expression for $F^N_{\Delta H}$ for $0 \leq P \leq 1.0$ represents the effect on radial power shapes of the control rods at the insertion limits.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of the full-length control rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. (5) Rod withdrawal block and load runback occurs if reactor trip set points are approached within a fixed limit. (6)

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient

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(2) FSAR Appendix 3A Section 5.3
(3) FSAR Appendix 3A Section 4.3

(5) FSAR Section 14.1.3
(6) FSAR Section 7.2.2

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 60% power, an automatic reactor trip will occur if any pump is lost or deenergized. (9) This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation and the overtemperature $\Delta T$ trip setpoint is adjusted to the value specified for four loop operation. When the overtemperature $\Delta T$ trip setpoint is adjusted to the value specified for three loop operation and the loop stop valve is closed in the inactive loop, the trip at 75% power will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation.

(9) FSAR Section 14.1.6

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

| Reactor Trip Channel Description Per Unit | 1. No. Of Channels | 2. No. Of Channels To Trip | 3. Minimum Operable Channels+++ | 4. Minimum Degree Of Redundancy+++ | 5. Operator Action If Column 3 or 4 Can Not Be Met + | 6. Setpoint++ |
|---|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 2 | 1 | 1 | 0 | Maintain Hot Shutdown** | N.A. |
| 2. Power Range High Flux (low set point)-interlocked with P-10 | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 25% of Rated Neutron Flux |
| 3. Power Range High Flux (high set point) | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 109% of Rated Neutron Flux |
| 4. Power Range High Flux Rate | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 15% of Rated Neutron Flux/5 sec. |
| 5. Negative Power Range Flux Rate | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 15% of Rated Neutron Flux/5 sec. |
| 6. Source Range Neutron Flux-Interlocked with P-10 and P-6 | 2 | 1 | 1 | 0 | Maintain Hot Shutdown*** (or CSD if that condition exists) | $10^5$ counts/sec. |
| 7. Intermediate Range Neutron Flux-Interlocked with P-10 | 2 | 1 | 1 | 0 | Maintain Hot Shutdown* | 25% of Rated Neutron Flux |
| 8. Overtemperature $\Delta T$, 4 loops | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | Actual $\Delta T \geq$ Programmed |
| Overtemperature $\Delta T$, 3 loops | 3 | 2 | 3 | 1 | Maintain Hot Shutdown** | Setpoint |
| 9. Overpower $\Delta T$, 4 loops | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | Actual $\Delta T \geq$ Programmed |
| Overpower $\Delta T$, 3 loops | 3 | 2 | 3 | 1 | Maintain Hot Shutdown** | Setpoint |
| 10. Pressurizer Low Pressure - interlocked with P-7 | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 1825 psig |
| 11. Pressurizer High Pressure | 4 | 2 | 3 | 2 | Maintain Hot Shutdown** | 2385 psig |

TABLE 3.1-1

Reactor Protection System - Limiting Operation Conditions and Setpoints



Amendment Nos. 62 & 59

| Reactor Trip Channel Description Per Unit | 1. No. Of Channels | 2. No. Of Channels To Trip | 3. Minimum Operable Channels*** | 4. Minimum Degree Of Redundancy*** | 5. Operator Action If Column 3 or 4 Can Not Be Met * | 6. Setpoint ** |
|---|---|---|---|---|---|---|
| 12. Pressurizer High Level - interlocked with P-7 | 3 | 2 | 2 | 1 | Maintain Hot Shutdown** | 92% of Span |
| 13. Low Primary Coolant Flow - interlocked with a. P-7 | 3 per loop | 2 per loop in 2 loops | 2 per loop | 1 | Maintain Hot Shutdown** | 90% of nominal span |
| b. P-8 | 3 per loop | 2 per loop in 1 loop | 2 per loop | 1 | Maintain Hot Shutdown** | |
| 14. RCP Bus Undervoltage - interlocked with P-7 | 1 per bus | 2 | 3 | 2 | Maintain Hot Shutdown** | 75% of nominal voltage |
| 15. RCP Bus Underfrequency - interlocked with P-7 and P-8 | 1 per bus | 2 | 3 | 2 | Maintain Hot Shutdown** | 57.0 Hz |
| 16. RCP Breaker Trip - interlocked with: | | | | | | |
| a. P-7, 4 loops | 4 breakers | 2 | 3 | 2 | Maintain Hot Shutdown** | N.A. |
| P-7, 3 loops | 3 breakers | 1 | 3 | 3 | Maintain Hot Shutdown** | N.A. |
| b. P-8, 4 loops | 4 breakers | 1 | 3 | 2 | Maintain Hot Shutdown** | N.A. |
| P-8, 3 loops | 3 breakers | 0 | NA | NA | Maintain Hot Shutdown** | N.A. |
| 17. Low Steam Generator Level in coincidence with feed flow-steam flow mismatch | 2 per loop | 1 per loop | 1 | 0 | Maintain Hot Shutdown** | 25% of narrow level range span |
| | 2 per loop | 1 per loop | 1 | 0 | Maintain Hot Shutdown** | and 0.7 x $10^6$ lbs/hr mismatch |
| 18. Low-Low Steam Generator Level - interlocked with loop isolation valve position | 3 per loop | 2 per loop | 2 per loop | 1 | Maintain Hot Shutdown** | 10% of narrow level range span |
| 19. Safety Injection | 2 | 1 | 2 | 1 | Maintain Hot Shutdown** | Any safety injection actuation |
| 20. Turbine Trip-interlocked with P-7 | 3 | 2 | 2 | 1 | Maintain Hot Shutdown** | N.A. |
| 21. Automatic Reactor Trip Logic | 2 | 1 | 2 | 1 | Maintain Hot Shutdown** | N.A. |

TABLE 3.1-1 (Continued)

Reactor Protection System - Limiting Operating Conditions and Setpoints

| PERMISSIVES | SETPOINT+++ |
|---|---|
| P-6 | $10^{-10}$ amps |
| P-7 | 10% Rated Neutron Flux and Pressure Equivalent to 10% of Rated Turbine Load |
| P-8 | 60% of Rated Neutron Flux (4 loops)<br>75% of Rated Neutron Flux (3 loops) |
| P-10 | 10% of Rated Neutron Flux |

+ If minimum conditions are not met within 24 hours, the unit shall be in the cold shutdown condition within an additional 24 hours.

++ Setpoints are ± established tolerances for instrument channel and setpoint errors as specified in "Channel Accuracies, Overall Channel Accuracies and Set-point Tolerances for W NES Process I and C Reactor Protection and Control Systems" August 30, 1971 - CEW-2652. The instruments shall not be set to exceed a Limiting Safety System Setting.

+++ For channel test, calibration or maintenance, the nimimum number of operable channels may be reduced by one but to not less than one, and the minimum degree of redundancy may be reduced by one but to not less than zero, for a maximum of two hours.

* When block conditions exist, maintain normal operation.

** 'Maintain Hot Shutdown' means maintain or proceed to hot shutdown within four hours if the unacceptable condition arises during power operation.

*** Verify shutdown margin immediately and comply with Section 3.2.1.A,B. When blocked conditions exist, maintain normal operation.

TABLE 3.1-1 (Continued)

Reactor Protection System - Limiting Operation Conditions and Setpoints

| | Reactor Trip Channel Description | Channel Check | Channel Calibration | Channel Function Test | Remarks |
|---|---|---|---|---|---|
| 17. | Low Steam Generator Level in Coincidence with Feed Flow Steam Flow Mismatch | S | R | M | |
| 18. | Low-Low Steam Generator Level | S | R | M | |
| 19. | Safety Injection | N.A. | N.A. | M(1) | (1) Manual SI function check at R only |
| 20. | Turbine Trip | N.A. | N.A. | M | |
| 21. | Automatic Reactor Trip Logic | N.A. | N.A. | M(1) | (1) Including Reactor Trip Breaker Opening |
| | PERMISSIVES | | | | |
| 22. | P-6 | N.A. | N.A. | S/U(1) | (1) Not required if performed within the previous seven days. |
| 23. | P-7 | N.A. | N.A. | M | |
| 24. | P-8 | N.A. | N.A. | M | |
| 25. | P-10 | N.A. | N.A. | M | |

NOTE: Specified intervals may be adjusted per Definition N, page 6.

S - Once Per Shift
D - Once Per Day
M - Once Per Month
R - Once Per Refueling Shutdown - calibration of these instruments may be done as much as six months prior to the start of refueling outage and still satisfy this requirement. The time between surveillances shall not exceed 20 months.

EFPM - Once Per Effective Full Power Month
Q - Once Per Quarter
S/U - Prior to Startup

N.A. - Not Applicable

TABLE 4.1-1 (Sheet 2 of 2)

Reactor Protection System Testing and Calibration Requirements*

*Applies to Unit I and Unit II

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | 4.2.1.B once a shift while remaining in this condition. During heatup, the boron concentration in the reactor coolant loops and pressurizer shall be sampled every 4 hours. The reactor coolant loop boron concentration must not decrease by more than 50 ppm between successive 4 hour samples. The pressurizer boron concentration must not be more than 200 ppm less than the reactor coolant loop boron concentration. |
| C. Unit Startup | C. Startup |
| 1. Immediately prior to startup, the reactor coolant temperature shall be shown to be greater than the temperature above which the moderator temperature coefficient is always negative (except during low power physics tests) and greater than 500 °F. | 1. The Tavg of each reactor coolant loop shall be logged before attempting to bring a reactor critical. |
| 2. When a reactor is approaching criticality, the shutdown banks shall be fully withdrawn in sequence (shutdown bank A,B,C,D) before any other rods are withdrawn. The control group rods shall be no further inserted than the limits shown by Figure 3.2-2 for Unit 1 and Figure 3.2-4 for Unit 2 for 4-loop operation and Figure 3.2-3 for Unit 1 and Figure 3.2-5 for Unit 2 for 3-loop operation when criticality is attained. | 2. Not Applicable. |
| D. Power Operation | D. Power Operation |
| 1. When a reactor is critical, except for physics tests and control rod exercises, the shutdown rods shall be fully with- | 1. Rod operation shall be verified by partial movement of all rods every two weeks. Rods which have been exercised within the |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.1.D.1. drawn and the control group rods shall be no further inserted than the limits shown on Figure 3.2-2 for Unit 1 and Figure 3.2-4 for Unit 2 for 4-loop operation and Figure 3.2-3 for Unit 1 and Figure 3.2-5 for Unit 2 for 3-loop operation. | 4.2.1.D.1. past two weeks during normal operation need not be verified Control rod bank positions with respect to its insertion limit shall be verified once per shift. |
| 2. Control bank insertion may be further restricted if the measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown as shown in Figure 3.2-1. | 2. Control rod bank worths shall be measured following each refueling outage. |
| 3. During physics tests and control rod exercises, the insertion limits need not be observed, but the limits in Figure 3.2-1 must be observed except during the low power physics test to determine total control rod worth and shutdown margin. For this test the reactor may be critical with all full length control rods fully inserted, except for the predicted most reactive rod. | 3. Not applicable. |
| 4. Three reactor coolant pumps per unit shall be operating whenever a reactor is critical except during natural circulation test, (power < 8% full power) or low power physics testing. | 4. Prior to proceeding from hot shutdown to hot standby, verify that three reactor coolant pumps are operating except during natural circulation tests or low power physics testing. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.1.D 5. Reactor power shall not be increased above 60% of rated power with only three reactor coolant pumps in operation unless the overtemperature $\Delta T$ trip setpoint and the P-8 interlock for three loop operation has been set in accordance with specification 2.1.1.B.4. | 4.2.1.D 5. Not Applicable |
| E. Rod Bank Assignment<br><br>Rod Bank Assignment shall be as delineated in Figure 3.2-8. Except during physics tests, the sequence of withdrawal of the control banks, when going from zero to 100% power, is A, B, C, D with control bank overlap. | E. Rod Bank Assignment<br><br>Rod Bank Assignment shall be verified after each refueling outage, for the refueled unit. |
| F. Boric Acid System (per unit)<br><br>1. A reactor shall not be taken from hot shutdown to hot standby unless the following conditions exist:<br><br>a. One boric acid tank for that reactor contains at least 5140 gallons of 11.5% (but not greater than 13%) by weight boric acid solution at a temperature of at least 145°F. | F. Boric Acid System (per unit)<br><br>1. Surveillance and testing of the boric acid system shall be performed as follows:<br><br>a. Boric acid tank level, concentration and temperature shall be verified prior to startup and weekly thereafter. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.1 G. Primary System Boron Concentration Changes during Cold Shutdown<br><br>When a boration or dilution operation is in progress, at least one reactor coolant pump or one residual heat removal loop shall be operating. | 4.2.1 G. Primary System Boron Concentration Changes during Cold Shutdown<br><br>The operation of at least one reactor coolant pump or one residual heat removal loop shall be verified before the start of a boration or dilution operation. |
| H. Reactivity Anomalies<br><br>A normalization of the computed boron concentration as a function of burnup shall be compared with the actual boron concentration of the coolant. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, the NRC shall be notified within 24 hours and an evaluation as to the cause of the discrepancy shall be made and reported to the NRC within 30 days. | H. Reactivity Anomalies<br><br>Reactivity anomaly evaluations shall be performed following startups after shutdowns of 72 hours or longer duration but shall not be required more than once if more than one such shutdown occurs in a two month period. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.2.A<br>2.2.c (Continued)<br><br>(1) Power between the maximum and minimum limits specified in 3.2.2.A.2.2.a.<br><br>(2) $\Delta$I within the $\Delta$I target band as per Section 3.2.2.A.4 and 3.2.2.A.5, except use ±3% $\Delta$I target band instead of the +6, -7% $\Delta$I target band.<br><br>2.2.d If any of the requirements of Section 3.2.2.A.2.2.c. are not maintained then power must immediately be reduced to below the power limited by APDMS type surveillance (Section 3.2.2.A.2.1.) and APDMS type surveillance must be initiated if the power is above $P_T$.<br><br>3.2.2.A.3<br><br>The target flux difference at a given power level, $P_O$, is determined by noting the indicated axial flux difference at the power level with equilibrium xenon conditions established in the core and with the full length rod bank more than 190 steps withdrawn. $P_O$ for the purpose of determining the target value, should be as high a power level as practicable. The target flux difference at any other level, P, is equal to the target value of $P_O$ multipled by the ratio, $P/P_O$. | 4.2.2.A<br>2.2.c.<br><br>(2) A flux difference alarm shall indicate non-conformance with the ±3% $\Delta$I target band for BASE LOAD operation. If the alarm is temporarily out of service, conformance with the applicable limit and the flux difference shall be logged hourly for the first 24 hours and half-hourly thereafter.<br><br>2.2.d Not Applicable<br><br>4.2.2.A.3<br><br>The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be determined at least once per equivalent full power quarter. The target difference should be updated every effective full power month. This may be done using the measured value for that month or by linear extrapolation using the two most recent measured values. The initial target flux difference on a reload may be determined from design predictions. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.2.A 8. For the purpose of determining penalties associated with deviations from the target band, time for use in applying Items 6.1 and 7.2 above shall be accumulated in the following manner:<br><br>8.1 For deviations at or below 50% power, time shall be accumulated such that a 1 minute actual deviation equals a 1/2 minute accumulative penalty in applying Items 6.1 and 7.2 above.<br><br>8.2 For deviations above 50% power, time shall be accumulated in a 1 for 1 time basis in applying Items 6.1 and 7.2 above. | 4.2.2.A 8. Not applicable. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.2 B. Quadrant Power Tilt Ratio Limits | 4.2.2 B. Quadrant Power Tilt Ratio |
| 1. If an indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, then within 2 hours, one of the following steps shall be taken:<br>a. Correct the tilt, or<br>b. Determine by measurement the core peaking factors and apply Specification 3.2.2.A, or<br>c. Restrict core power level so as not to exceed full rating less 3% for each percent of quadrant power tilt ratio beyond 1.0. | 1. Quadrant power tilt ratio shall be calculated and logged along with the individual upper and lower excore calibrated outputs as follows:<br>a. Once each shift at power levels greater than 50%.<br>b. Four times a shift and following a load change of more than 10% power at any power level above 50% if one or both quadrant power tilt alarms are inoperable. |
| 2. If an indicated quadrant power tilt ratio exceeds 1.02 for a period of 24 hours without known cause, or if sudden tilt reoccurs intermittently without known cause, the reactor shall be put in the Hot Shutdown Condition within 8 hours. However, operation below 50% of rated power, for testing and/or correcting the tilt, shall be permitted. | 2. Not Applicable |

LIMITING CONDITION FOR OPERATION

3.2.2.B 3. If an indicated quadrant power tilt exceeds 1.09, except for physics testing, the reactor shall be put in the Hot Shutdown Condition; however, operation below 50% of rated power, for testing and/or correcting the tilt, shall be permitted.

C. Instrumentation (per unit)

1. Excore axial imbalance detector system

a. The excore axial imbalance detector system shall be recalibrated at least every three effective full power months. The calibration shall be checked each effective full power month using the INCORE SYSTEM and recalibrated if the difference is > 1%. The minimum requirements per flux map used for the recalibration are:

1. At least 16 different thimble traces, and

2. At least 2 differenct thimble traces, per quadrant.

b. If requirement 3.2.2.C.1.a cannot be met, then power shall be limited to 90% of rated power for 4 loop operation and 60% of rated power for 3 loop operation.

SURVEILLANCE REQUIREMENT

4.2.2.B 3. Not Applicable

C. Instrumentation (per unit)

1. Excore axial imbalance detector system

a. Not Applicable

b. Not Applicable

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.2.C 2. Inoperable Excore Detector Channel<br>If an excore detector channel is inoperable, quadrant power tilt ratio shall be determined by periodically monitoring incore thermocouples. | 4.2.2.C.2. Inoperable Excore Detector Channel<br>If an excore detector channel is inoperable, quadrant power tilt ratio shall be determined by monitoring at least four thermocouples per quadrant once an hour and after any load change greater than 10% at any power level above 50%. |
| 3. NIS Detector Temperature Control | 3. NIS Detector Temperature Control |
| a. One of the two reactor cavity ventilation fans (Unit I 1RV012-1A, 1RV013-1B) or (Unit II 2RV012-2A, 2RV013-2B) shall be operating whenever Tavg is greater than 145°F. | a. Reactor cavity ventilation fan operation shall be verified once a shift. |
| b. If this condition cannot be met, the reactor shall be brought to the Hot Shutdown Condition immediately. | b. Not Applicable. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2 3. Control Rod System Operability (per unit) | 4.2 3. Control Rod System Operability (per unit) |
| A. Rod Misalignment Limitations | A. Rod Misalignment Limitations |
| 1. If a full-length control rod is out of alignment with its bank by more than ±12 steps indicated; then, within 2 hours, one of the following steps shall be taken: | 1. A rod malposition check shall be made once a shift using both the analog and digital displays. |
| a. Realign the rod, or | |
| b. Determine by measurement the core peaking factors and apply Specification 3.2.2.A, or | |
| c. Restrict power level to 80% of rated power. | |
| 2. If the misaligned control rod is not realigned within 8 hours, the rod shall be declared inoperable and the limitations of 3.2.3.B apply. | 2. Not Applicable |
| 3. The provisions of specifications 3.2.3.A and 3.2.3.B shall not apply during physics tests in which the control rods are intentionally misaligned. | 3. Not Applicable |
| B. Inoperable Rod Limitations | B. Inoperable Rod Limitations |
| 1. An inoperable control rod is a rod which cannot be moved by its mechanism or which is declared inoperable by Specification 3.2.3.A or 3.2.3.C. | 1. Not Applicable |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.3.B 2. Not more than one inoperable control rod shall be permitted during power operation. If more than one rod is determined to be inoperable, the reactor shall be placed in the hot shutdown condition within 4 hours. | 4.2.3.B 2. Not Applicable. |
| 3. If more than one control rod is inoperable because of a Rod Urgent Failure in the rod control system, the provisions of Specifications 3.2.3.B.1 and 3.2.3.B.2 above shall not apply. If the affected assemblies cannot be returned to service within two hours, the reactor shall then be placed in the hot shutdown condition within 4 hours. | 3. Not Applicable. |
| 4. Deleted | 4. Deleted |
| 5. If an inoperable full-length rod is located above the 200 step level and is capable of being tripped, then the insertion limits in Figure 3.2-2 for Unit I and Figure 3.2-4 for Unit II shall apply for 4 loop operation and the insertion limits in Figure 3.2-3 for Unit I and Figure 3.2-5 for Unit II shall apply for 3 loop operation. | 5. Not Applicable. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.3 | 4.2.3 |
| C. Rod Drop Time | C. Rod Drop Time |
| The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 1.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: | Under the conditions of LCO 3.2.3.C.1 and 2.2.3.C.2 the hot rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality: |
| 1. $T_{avg} \geq 530^{\circ}F$, and | 1. For all rods following each removal of the reactor vessel head, |
| 2. All reactor coolant pumps operating. | 2. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which would affect the drop time of those specific rods, and |
| APPLICABILITY: Applies to all operation with the reactor critical. | 3. At least once per refueling outage (not to exceed 20 months). |
| ACTION: With the drop time of any full length rod determined by Surveillance Requirement 4.2.3, to exceed the above limit restore the rod drop time to within the above limit or declare the rod inoperable per Section 3.2.3A prior to proceeding to criticality. | |
| D. Inoperable Rod Position Indicator Channels | D. Inoperable Rod Position Indicator Channels |
| 1. Not more than one rod position indicator channel per control rod group nor two rod position indicator channels per control rod bank shall be permitted to be inoperable at any time, except during hot rod drop timing measurements. | 1. If a rod position indicator is out of service, then: a. For operation between 50% and 100% of rated power, the position of the control rod |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.2.3.D.1 2. If the conditions of Section 3.2.3.D.1 cannot be met the reactor shall be brought to at least the Hot Shutdown condition within four hours and the reactor trip breakers shall remain open. | 4.2.3.D.1.a shall be checked indirectly by excore detectors and/or thermocouples and/or moveable incore detectors every shift; or after any rod motion of the non-indicating rod, exceeding 12 steps, whichever occurs first. |
| | b. During operation below 50% of rated power, no special monitoring is required. |
| 4. DNB Parameters<br><br>A. The following DNB related parameters shall be maintained within the limits shown during operation.<br><br>1. Reactor Coolant System Tavg<br>FOUR LOOP: ≦ 566.3°F<br>THREE LOOP: #<br>2. Pressurizer Pressure<br>FOUR LOOP: ≧ 2220 psia (2205 psig)*<br>THREE LOOP: #<br>3. Reactor Coolant System Total Flow Rate<br>FOUR LOOP: ≧ 356,000 GPM<br>THREE LOOP: # | 4.A.1. Each of the parameters listed in Specification 3.2.4.A shall be verified to be within its limit at least once per 12 hours.<br><br>2. The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months. |
| B. With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power within the next 4 hours. | |

* Limit not applicable during either a thermal power ramp increase in excess of 5% rated thermal power per minute or a thermal power step increase in excess of 10% rated thermal power.

\# Parameter limits for three loop operation to be established prior to operation above P-7 with less than four loops operating.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ (1), there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) abnormal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting $F_Q$, (b) the operator has a direct influence on $F_Q$ through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in $F_Q$ by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation it is not necessary to measure these quantities continuously. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of $\pm$ 12 steps, not including instrument error, precludes a rod misalignment no greater than 15 inches. With maximum instrumentation error considered the actual rod misalignment is no more than 24 steps or 15 inches.

2. Control rod banks are sequences with overlapping banks as described in Technical Specification 3.2.

3. The full length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux differences control or additional axial power monitoring and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In Specifications 3.2.2, $F_Q$ is arbitrarily limited for $P \leq 0.5$.

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during loadfollow maneuvers. Basically control of flux difference is required to limit the difference between the current value of Flux Difference ($\Delta$I) and a reference value which corresponds to the full power equilibrium value of Axial offset (Axial Offset = $\Delta$I/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the $F_Q$ limit is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference ) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating, is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of the $\Delta$I target band are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. The alarms provided are derived from the plant process computer which determines the one minute averages of the operable excore detector outputs to monitor $\Delta$I in the reactor core and alerts the operator when $\Delta$I alarm conditions exist. Two types of alarm messages are output. Above a preset power level, an alarm message is output immediately upon determining a delta flux exceeding a preset band about a target delta flux value. Below this preset power level, an alarm message is output if the $\Delta$I exceeded its allowable limits for a preset cumulative amount of time in the past 24 hours. For periods during which the alarm on flux difference is inoperable, manual surveillance will be utilized to provide adequate warning of significant variations in expected flux differences. However every attempt should be made to restore the alarm to an operable condition as soon as possible. Any deviations from the target band during manual logging shall be treated as deviations during the entire preceeding logging interval and approprate actions shall be taken. This action is necessary to satisfy NRC requirements; however more frequent readings may be logged to minimize the penalty associated with a deviation from the target band to justify continued operation at the current power.

The times that deviations from the band occur are normally accumulated by the computer.

significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range $\pm(\gamma+1)$ percent ($+\gamma$ percent to $-\gamma$ percent indicated) increasing by $\pm1$ percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90% of $P_T$ or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\Delta I$ target band for as long a period as one hour, then xenon distribution may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events, the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of transients considered.

In accordance with the approved Westinghouse model as presented in WCAP-8381, no collapses are expected throughout the fuel cycle of operation. The predicted minimum times for clad flattening are:

| Fuel Region | EFPH |
|---|---|
| 1 | 19,500 |
| 2 | > 30,000 |
| 3* | > 30,000 |

for Zion Unit 1. The predicted minimum times to collapse for Unit 2 are:

| Fuel Region | EFPH |
|---|---|
| 1 | 27,000 |
| 2 | > 30,000 |
| 3* | > 30,000 |

A design criterium requires that proposed reload fuel region exposure levels expected at the time of discharge not exceed the predicted minimum collapse time. Operation in the exposure range in which clad collapse is postulated is not permitted under these technical specifications.

The predicted minimum time to collapse for all reload fuel regions is greater than 30,000 effective full power hours.

*Except that the four (4) Region 3 assebmlies to be used in the Extended Burnup Program for Zion Unit 2 have a predicted minimum time to collapse greater than 41000 EFPH.

DNB Parameters: The limits on the DNB related paramters assure that each of the parameters is maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

A quadrant power tilt will be indicated by the excore detectors by the arrangement of the current recorders on the control board. Four 2-pen recorders are provided, the pens are grouped so that, in the absence of a quadrant power tilt, the two

-ink traces coincide. Any divergence in the traces indicate a power tilt. Furthermore, a quadrant power tilt alarm is provided for the upper and lower sets of excore currents.

Quadrant power tilt ratio limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt ratio limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. (8) Operational experience shows that normal power tilt ratios are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could buildup. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The quadrant power tilt ratio of 1.02 at which remedial and corrective action is required has been set so as to provide DNB and linear heat generation rate protection with x-y power tilts. Analyses have shown that fractional increases in the x-y power peaking factor are less than or equal to twice the increase in the indicated quadrant power tilt ratio, i.e., an envelope with a 2:1 slope.

As described above, an uncertainty factor of 1.08 is included in $F_{\Delta H}^N$ and 1.05 in $F_Q^N$. Therefore, a limiting power tilt ratio of 1.025 can be tolerated before the margin for uncertainty in $F_Q^N$ is depleted. However, a measurement uncertainty is associated with the indicated quadrant power tilt ratio. Thus, allowing for a low measurement of power tilt ratio, the action level of indicated tilt ratio has been set at 1.02. An alarm is set to alert the operator to an indicated tilt ratio of 1.02 or greater and that action is required. To avoid unnecessary power changes, the operator is allowed two hours in which to verify with in-core mappings and/or to determine and correct the cause of the tilt. Should this action not be taken, the margin for uncertainty in $F_Q^N$ is reinstated by reducing the power by 2 percent for each percent of tilt ratio above 1.0, in accord with the 2:1 slope envelope described above, or as required by the restriction on peaking factors.

The upper limit on the quadrant tilt ratio at which hot shutdown is required has been set at 1.09 so as to provide protection against excessive linear heat generation rate.

The nuclear ion chambers located outside a reactor vessel measure the flux distribution of the top and bottom halves of a core. Core traverses in a few of the in-core instrument thimbles will establish that the excore flux measurement equipment is properly calibrated. Operating experience has established that the excore flux measurement system is of a reliable design, and that the 10% load reduction, in the event of a recalibration delay, is an ultra conservative compensation.

Operating experience at similar PWR plants has shown that quadrant power tilts determined by monitoring symmetric thermocouples are in very good agreement with quadrant power tilts determined from power distribution maps using the Movable Detector System.

Operation of one reactor cavity vent fan per unit ensures an adequate flow rate of cooling air to each NIS Detector (9).

The various control rod assemblies (shutdown banks, control banks A, B, C, D) are each to be moved as a bank, that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position (10). The rod position indicator channel is sufficiently accurate to detect a misaligned rod 15 inches away from the demand position of the bank. The indicated $\pm$ 12 step permissible misalignment provides an enforceable limit below which design distribution is not exceeded.

In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors.

One inoperable control rod per unit is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the case analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 3 day period allowed for the analysis is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The rod drop time of 1.8 seconds is based on the negative reactivity insertion rate used in accident analysis. (11)


(1) FSAR - Figure 3.2.1-8
(2) FSAR - Table 3.2.1-1
(3) FSAR - Figure 3.2.1-11
(4) FSAR - Chapter 14
(5) FSAR - Section 3.1.2
(6) FSAR - Section 3.1.3
(7) FSAR - Chapter 14, Appendix C
(8) FSAR - Question 3.8
(9) FSAR - Section 9.10.6
(10) FSAR - Section 7.3
(11) FSAR - Figure 14-2
(12) August 27, 1976 Order for Modification of License.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | 4.3.1.B.5 <u>Reports</u><br><br>A. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.<br><br>B. The complete results of the steam generator tube inservice inspection shall be included in the Special Report pursuant to Specification 6.6.3.(c). The Special Report shall be submitted within 12 months following completion of the inspection. This report shall include:<br><br>1. Number and extent of tubes inspected.<br><br>2. Location and percent of wall thickness penetration for each indication of an imperfection.<br><br>3. Identification of tubes plugged.<br><br>C. Results of steam generator tube inspections which fall into category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.6.2 prior to resumption of plant operation. The written follow up of this report shall provide a description of investigation conducted to determine |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.3.2 Pressurization and System Integrity | 4.3.2 Pressurization and System Integrity |
| A. Heatup and Cooldown | A. Not Applicable. |

3.3.2 Pressurization and System Integrity

A. Heatup and Cooldown

The Reactor Coolant System temperature and pressure (with the exception of the pressurizer) shall be limited in accordance with the limit lines shown in Figures 3.3.2-1 and 3.3.2.2 during heatup, cooldown and inservice leak and hydrostatic testing with:

1. a. A maximum heatup of 100°F in any one hour period.

   b. A maximum cooldown of 100°F in any one hour period.

   c. A maximum temperature change of $\leq$10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2. Figures 3.3.2-1 and 3.3.2-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

3. Allowable combinations of pressure and temperature for specified temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
| --- | --- |
| 3.3.2 (Continued) | 4.3.2 |
| B. The limit lines shown in Figures 3.3.2-1 and 3.3.2-2 shall be recalculated periodically as required, based on results from the material surveillance program. | B. Not Applicable |
| C. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the primary and secondary coolant is below 70°F. | C. Not Applicable |
| D. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F. | D. Not Applicable |
| E. Hydrostatic Testing<br>1. System inservice leak and hydrotests shall be performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition, up to and including Summer 1975 Addendum. | E. Not Applicable |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.3.2. F. Safety Injection Actuation<br><br>1. If safety injection should occur when a reactor is in the hot shutdown condition or above, the reactor shall remain in the hot shutdown condition until the status of the reactor coolant system integrity is determined.<br><br>2. If the inspection and review (Sec. 6.1.G.2.a(7) and Sec. 6.3) of the reactor coolant system integrity determines that:<br><br>a. The injection did not affect reactor coolant system integrity the plant may proceed to power operation.<br><br>b. The injection did affect reactor coolant system integrity, the reactor shall be placed in the cold shutdown condition within 24 hours.<br><br>3. In the event the ECCS is actuated and injects water into the Reactor Coolant System when Tavg $\geq 350^\circ F$, a Special Report shall be prepared and submitted to the Commission within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current valve of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its valve exceeds 0.70. | 4.3.2 F. Not Applicable |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.3 3. Leakage (per unit)<br><br>A. If the leakage rate, from other than controlled leakage sources, such as the Reactor Coolant Pump Controlled Leakage Seals, exceeds 1 gpm and the source of the leakage is not identified within twenty-four hours of detection, the reactor shall be brought to hot shutdown within four hours. If the source of the leakage is not identified within an additional 24 hours, the reactor shall be brought to a cold shutdown condition within 24 hours. | 4.3 3. Leakage (per unit)<br><br>A. When Reactor Coolant System pressure is greater than 500 psig, one of the following monitoring requirements shall be performed (4.3.3.A.1 or 4.3.3.A.2):<br><br>1. Containment activity shall be continuously monitored by radiation detectors RE-0011A or RE-0012A.<br><br>2. Manual sampling of the containment atmosphere shall be performed once a shift |
| B. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in 3.3.3.C below. | B. When Reactor Coolant System pressure is greater than 500 psig at least three of the following monitoring requirements shall be performed (4.3.3.B.1, 2, 3, 4, and 5): |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.3.3.B | 4.3.3.B 1. The amount of Reactor Coolant System makeup water required to maintain pressurizer level and volume control tank level shall be recorded.<br>2. Containment sump and reactor cavity sump water accumulation shall be monitored daily.<br>3. Containment pressure, temperature and humidity shall be monitored.<br>4. The high temperature alarm (TE-401) in the reactor head flange leakoff piping shall be operable.<br>5. The Reactor Vessel Leak Detection system (RE-PR12A, RE-PR12B, RY-PR12A, and associated alarms) shall be operable. |
| C. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe wall, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition within 24 hours and corrective action taken prior to resumption of unit operation. | C. If the monitoring performed in sections 4.3.3.A and 4.3.3.B indicates significant leakage a detailed investigation shall be performed to identify the sources and quantity of leakage. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.3.4 | 4.3.4 D. Materials Irradiation Surveillance Specimen Inspection (per unit) |

Specimen capsules to be used in the reactor vessel material surveillance program shall be withdrawn during the refueling period either immediately preceeding or following the Effective Full Power Years (EFPY) of unit life as follows:

| Capsule Designation | Withdrawal Schedule (EFPY) |
|---|---|
| T or U | 1.2 |
| U or T | 3.3 |
| X | 5.8 |
| Y | 8.3 |
| W,S,V,Z | Standby |

Basis

4.3.4 The surveillance inspection program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). The design of the plant, state of non-destructive testing technology, and access to areas to be inspected require such relief.

The Reactor Vessel Material Surveillance Program is designed to evaluate the effects of radiation on the fracture toughness of reactor vessel steel based on the transition temperature approach and the fracture mechanics approach.

10 CFR 50, Appendix H, Paragraph II C.3.C requires that the Reactor Vessel Material Surveillance Program shall provide for the testing of at least five capsules with the following withdrawal schedule:

First Capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one/fourth service life, whichever is earlier.

Second and Third Capsules - At approximately one/third and two/thirds of the time interval between first and fourth capsule withdrawal.

Fourth Capsule - Three/fourths of service life.

Fifth Capsule - Standby.

The withdrawal schedule using EFPY's of unit life is based on a lead factor of 2.9, a capacity factor of approximately 80% and a service life of 40 years.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

## 3.7 STEAM GENERATOR EMERGENCY HEAT REMOVAL

Applicability:

Applies to auxiliary feedwater system and steam generator safety valves, per unit.

Objective:

To insure adequate plant cooldown capability upon loss of normal feedwater flow and loss of main condenser vacuum.

Specification:

1. Steam Line Safety Valves

   A. Twenty ASME code safety valves (5 per steam generator) shall be operable whenever the reactor is heated above 350°F except as specified in 3.7.1.C, 3.7.1.D, and 3.7.1.E.

   B. Deleted

## 4.7 STEAM GENERATOR EMERGENCY HEAT REMOVAL

Applicability:

Applies to surveillance of auxiliary feedwater system, and steam generator safety valves per unit.

Objective:

To insure availability of the above system and valves.

Specification:

1. Steam Line Safety Valves

   A. Ten steam generator safety valves per unit shall be tested for set pressure at each refueling outage. Testing shall be done by a calibrated auxiliary lifting device or by bench testing on compressed gas. At least two of the valves tested shall be from each orifice size ("Q" or "R"). All valves on a unit shall have been tested at the end of each second refueling outage. The valves and the corresponding set pressures and orifice sizes are identified in Table 4.7-1.

   B. Deleted

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | 4.8.4.B A flow balance test shall be performed on the affected lines during the next refueling outage following valve stroking or maintenance or other system modifications which might alter the E.C.C.S. flow characteristics.<br><br>E.C.C.S. flow rates for single pump operation shall meet the following requirements under the minimum resistance configuration:<br><br>1. Charging pump cold leg injection plus seal injection shall not exceed 550 GPM;<br><br>2. SI pump hot or cold leg injection plus mini flow shall not exceed 650 GPM;<br><br>3. The minimum charging pump cold leg injection through any 3 lines shall be 275 GPM; and<br><br>4. The minimum SI pump cold leg injection through any 3 lines shall be 400 GPM. |

| Component Name | Component Number |
| --- | --- |
| Residual Heat Removal Pump-1A (2A)<br>Residual Heat Removal Pump-1B (2B) | RH001-1A (2A)<br>RH002-1B (2B) |
| Residual Heat Exchanger-1A (2A)<br>Residual Heat Exchanger-1B (2B) | RH003-1A (2A)<br>RH004-1B (2B) |
| Recirculation Sump to RHR Pump Suction Valves | MOV-SI8811A<br>MOV-SI8811B |
| RWST to RHR Pump Suction Valves | MOV-RH8700A<br>MOV-RH8700B |
| Isolation Valves from Reactor Coolant System to RHR Pumps | MOV-RH8701<br>MOV-RH8702 |
| Residual Heat Removal Pumps' Suction Valves | MOV-SI8812A<br>MOV-SI8812B |

Residual Heat Removal Pump System

TABLE 4.8-3

All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

Fifty three full length rod cluster control assemblies consisting of 20 individual 80% Ag - 15% In - 5% Cd alloy stainless steel clad rods are inserted into the guide thimbles at appropriate locations in the core.

Burnable poison rods consisting of Borosilicate glass sealed in stainless steel tubes may be used for reactivity and/or power distribution control.

## 5.4 Containment System

### 5.4.1 Design Basis

The reactor containment completely encloses the entire Reactor Coolant System and assures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the Reactor Coolant System were to occur. The design of the containment liner with channels and the penetrations permits a much more sensitive and accurate means of testing the containment leakage status more frequently than is possible with a conventional integrated leak rate test. The structure provides biological shielding for both normal and Design Basis Accident situations. (1)

### 5.4.2 Containment System Structure

The Reactor Containment is in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The foundation slab is conventionally reinforced with high-strength reinforcing steel. The entire structure is lined with one-quarter inch welded steel plate to provide vapor tightness.

The approximate dimensions of the Reactor Containment are: inside diameter, 140 feet; inside height, 212 feet; vertical wall thickness, 3-1/2 feet; dome thickness 2'-8"; and the foundation slab thickness, 9 feet. The containment encloses the pressurized water reactor, steam generators, reactor coolant loops and portions of the auxiliary systems and engineered safeguards systems.

## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 Organization, Review, Investigation, and Audit.

A. The Station Superintendent shall have overall full-time responsibility for safe operations of the facility. During periods when the Station Superintendent is unavailable, he shall designate this responsibility to an established alternate who satisfies the ANSI N18.1 experience requirements for plant manager.

B. The organization chart of the corporate management which relates to the operation of this station and the normal functional organization chart for operation of the station is shown in Figures 6.1.1.

C. The shift manning for the station shall be as shown in Figure 6.1.2. The Operating Assistant Superintendent, Operating Engineer, Shift Engineers, and Shift Foreman shall have a senior operating license. The Fuel Handling Foreman has a limited Senior Operating License. The Division Vice President, Nuclear Stations on the corporate level has responsibility for the Fire Protection Program. An Operating Engineer at the station will be responsible for implementation of the Fire Protection Program. A Fire Brigrade of at least 5 emebers shall be maintained onsite at all times. The Fire Brigade shall not include the minimum shift crew necessary for safe shutdown of the plant (4 members) or any personnel required for other essential functions during a fire emergency.

D. Qualifications of the Station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1 "Selection and Training of Nuclear Power Plant Personnel," dated March 6, 1971 with the exception of the Rad-Chem Supervisor or Lead Health Physicist, who shall meet or exceed the qualifications of Radiation Protection Manager of Regulatory Guide 1.8 September, 1975. The individual filling the position of Administrative and Support Services Assistant Superintendent shall meet the minimum acceptable level for "Technical Manager" as described in 4.2.4 of ANSI N18.1, 1971.

E. Retraining and replacement training of Station personnel shall be accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. A training program for the Fire Brigade shall be maintained under the direction of the Station Fire Marshall and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975 except that Fire Brigade training will be conducted quarterly.

F. Retraining shall be conducted at intervals not exceeding two years.

G. The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below:

1. The Supervisor of the Offsite Review and Investigative Function shall be appointed by the Executive Vice-President (Construction, Production, and Engineering). The Audit Function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Supervisor of the Offsite Review and Investigative function shall: (i) provide directions for the review and and investigative function and appoint a senior participant to provide appropriate direction, (ii) select each participant for this function, (iii) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (iv) independently review and approve the findings and recommendations developed by personnel performing the [illegible]view and investigative function, (v) ap[illegible] and report in a timely manner all findi[illegible] noncompliance with NRC requirement[illegible] provide recommendations to the Station Superintendent, Division Vice President Nuclear Stations, Manager of Quality Assurance, Vice President (Nuclear Operations) and the Executive Vice-President (Construction, Production, and Engineering).

During the periods when the Supervisor of the Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate who satisfies the formal training and experience requirements for the supervisor of the Offsite Review and Investigative Function.

The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

(1) The safety evaluations for 1) changes to procedures, equipment or systems as described in the safety analysis report and 2) tests or experiments completed under the provision of 10 CFR Section 50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance.

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

(3) Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59 10 CFR.

(4) Proposed changes in Technical Specifications or NRC operating licenses.

Offsite Review and Investigative Function (Continued)

(5) Noncompliance with NRC requirements, or of internal procedures or instructions having nuclear safety significance.

(6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

(7) Reportable Occurrences requiring 24 hour notification to the Commission.

(8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

(9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change.

(10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Vice President-Nuclear Stations and Manager of Quality Assurance.

b. Audit Function

The Audit Function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance for Operating and to the Staff Assistant to the Manager of Quality Assurance for maintenance quality assurance activities.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within a period of two years or as designated below.

(1) Audit of the Conformance of facility operation to provisions contained within the Technical Specification and applicable license conditions at least once per year.

(2) Audit of the adherence to procedures, training and qualification of the station staff at least once per year.

(3) Audit of the results of actions taken to correct deficiencies occuring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

(4) Audit of the performance of activities required by the Quality Assurance Program to meet the Criteria of Appendix "B", 10 CFR 50.

(5) Audit of the Facility Emergency Plan and implementing procedures.

(6) Audit of the Facility Security Plan and implementing procedures.

(7) Audit onsite and offsite reviews.

(8) Audit the Facility Fire Protection Program and implementing procedures at least once per 24 months.

(9) An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

(10) An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

(11) Report all findings of noncompliance with NRC requirements and recommendations and results of each audit to the Station Superintendent, the Division Vice President-Nuclear Stations, Manager of Quality Assurance, Vice President (Nuclear Operations) Director of Nuclear Licensing, and to the Executive Vice President (Construction, Production, and Engineering).

c. Authority

The Manager of Quality Assurance reports to the Chairman and President and the Supervisor the Offsite Review and Investigative Function reports to Director of Nuclear Safety who reports to the Chairman and President. Either the Manager of Quality Assurance or the Supervisor of the Offsite Review and Investigative Function has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

d. Records

(1) Reviews, audits and recommendations shall be documented and distributed as covered in 6.1.G.1.a and 6.1.G.1.b.

(2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the off-site reviews and investigative functions described in Specifications 6.1.G.1.a. These procedures shall cover the following:

(1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees and consultants.

(3) Review and approval

(4) Detailed listing of items to be reviewed.

(5) Method of (a) appointing personnel, (b) performing reviews, investigations, (c) reporting findings and recommendations of reviews and investigations, (d) approving reports, and (e) distributing reports.

(6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

(1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor of the Offsite Review and Investigative Function, shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated.

(a) nuclear power plant technology
(b) reactor operations
(c) utility operations
(d) power plant design
(e) reactor engineering
(f) radiological safety
(g) reactor safety analysis
(h) instrumentation and control
(i) metallurgy
(j) any other appropriate disciplines required by unique characteristics of the facility.

(2) Individuals performing the Review and Investigative Function shall possess a minimum formal training and experience as listed below for each discipline.

(a) Nuclear Power Plant Technology

Engineering graduate or equivalent with 5 years experience in the nuclear power field design and/or operation.

B. 1. Procedures for items identified in Specification 6.2.A and any changes to such procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in the areas of operation and fuel handling, and by the Maintenance Assistant Superintendent and Technical Staff Supervisor in the areas of plant maintenance, instrument maintenance, and plant inspection. Procedures for items identified in Specification 6.2.B and any changes to such procedures shall be reviewed and approved by the Technical Staff Supervisor and the Rad-Chem Supervisor. At least one person approving each of the above procedures shall hold a valid senior operator's license. In addition, these procedures and changes thereto must have authorization by the Station Superintendent before being implemented.

2. Work and instructions type procedures which implement approved maintenance or modification procedures shall be approved and authorized by the Maintenance Assistant Superintendent where the written authority has been provided by the Station Superintendent. The "Maintenance/Modification Procedure" utilized for safety related work shall be so approved only if procedures referenced in the"Maintenance/Modification Procedure" have been approved as required by 6.2.A. Procedures which do not fall within the requirements of 6.2.A or 6.2.B may be approved by the Department Heads.

C. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

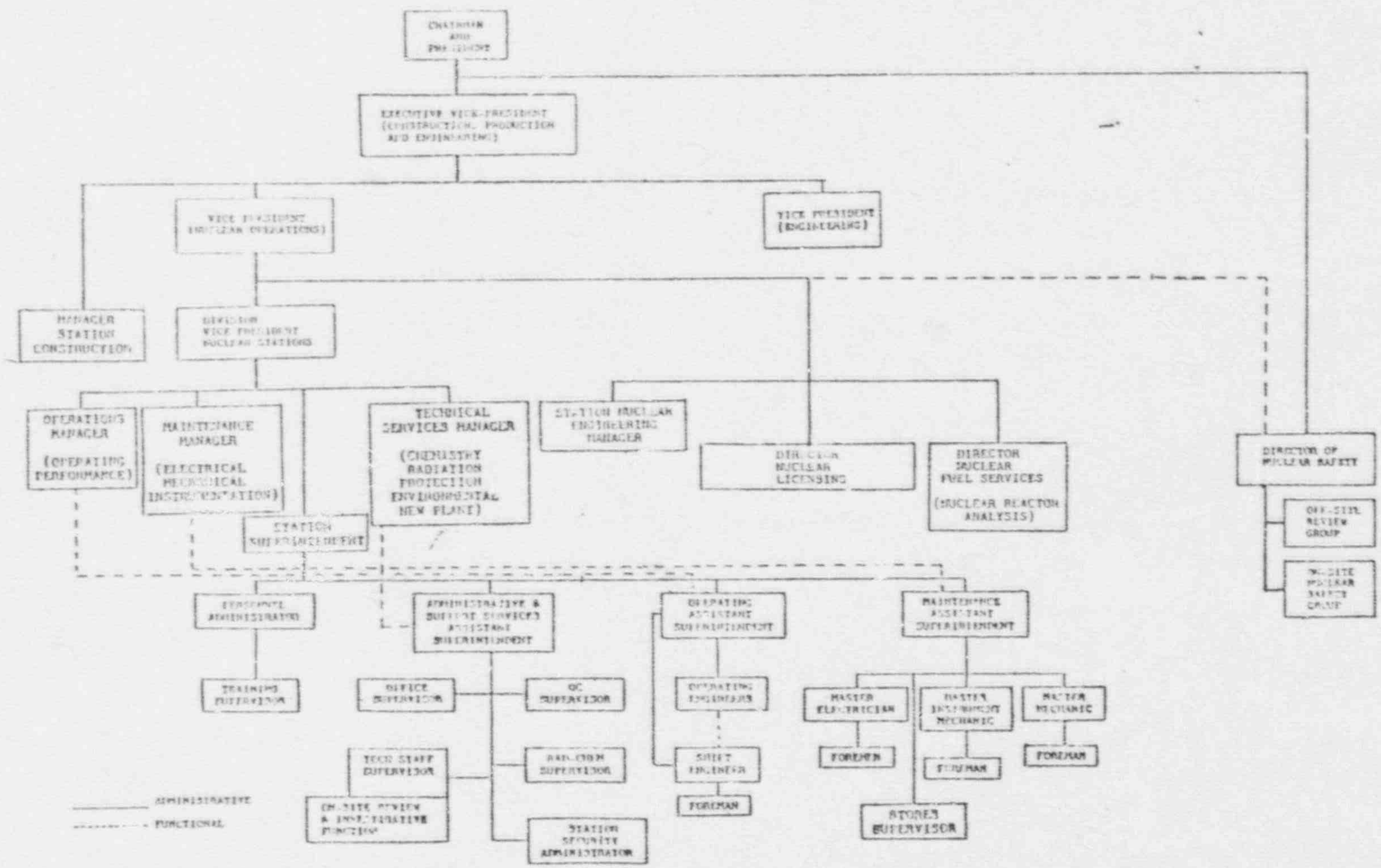
2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed by the Onsite Review and Investigative function and approved by the Station Superintendent within 14 days of implementation.

D. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted quarterly. These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

# CORPORATE AND ON-SITE ORGANIZATION

## Zion Station



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| Position | Unit in Operating Mode (other than cold shutdown) | | |
|---|---|---|---|
| | None | 1 or 2 | 1 & 2 |
| Shift Engineer<br>or<br>Shift Foreman | 1 | 1 | 2 |
| Nuclear Station Operator | 1 | 2 | 3 |
| Equipment Operator<br>or<br>Equipment Attendant | 2 | 3 | 4 |
| Rad-Chem Technician | 1 | 1 | 1 |
| TOTAL | 5 | 7 | 10 |
| MINIMUM* | 5 | 6 | 9 |

*The minimum number refers only for the case of shift shortage, caused by a sudden sickness or home emergency.

Notes:

1. SRO shall be present on site at all times when there is fuel in the reactor.

2. A licensed man shall be in the control room at all times whenever fuel is in either reactor.

3. Two licensed men shall be in the control room during reactor startups, shutdowns, operation, and other periods such as planned control rod manipulations.

4. For the period of Unit 1 and Unit 2 Start up Test Program, two licensed men per unit shall be in the control room during any operation of the reactor or plant which can cause changes in reactivity which have not been verified previously by the startup test program.

ZION SHIFT MANNING CHART
Figure 6.1.2

a. Onsite Review and Investigative Function (Continued)

by personnel performing the Review and Investigative Function; (v) report all findings of noncompliance with NRC requirements, and provide recommendations to the Division Vice President-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and (vi) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

The responsibilities of the personnel performing this function are stated below:

(1) Review of: 1) procedures required by Specification 6.2 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

(2) Review of all proposed tests and experiments that affect nuclear safety.

(3) Review of all proposed changes to the Technical Specifications.

(4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

(5) Investigation of all noncompliance with requirements and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Vice President-Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function.

(6) Review of facility operations to detect potential safety hazards.

(7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function.

(8) Review of the Station Security Plan and shall submit recommended changes to the Division Vice President-Nuclear Stations.

(9) Review of the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Vice President-Nuclear Stations.

(10) Review of reportable occurrences and actions taken to prevent recurrence.

b. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation and administrative procedures relating to facility

b. Authority (Continued)

operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Vice President-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

c. Records

(1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Vice President-Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent Function, the Station Superintendent and the Manager of Quality Assurance.

(2) Copies of all records and documentation shall be kept on file at the station.

d. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

(1) Content and method of submission and presentation to the Station Superintendent, Division Vice President-Ncuear Stations and the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees.

(3) Review and approval.

(4) Detailed listing of items to be reviewed.

(5) Procedures for administration of the quality control activities.

(6) Assignment of responsibilities.

e. Personnel

(1) The personnel performing the Onsite Review and Investigative Function, in addition to the Station Superintendent, shall consists of persons having expertise in:

(a) nuclear power plant technology
(b) reactor operations
(c) reactor engineering
(d) radiological safety and chemistry
(e) instrumentation and control
(f) mechanical and electric systems.

(2) Personnel performing the Onsite Review and Investigative Function shall meet minimum acceptable levels as described in ANSI N18.1 1971, Sections 4.2 and 4.4.

6.3 Action to be Taken in the Event of an Reportable Occurrence in Plant Operation

Any reportable occurrence shall be promptly reported to the Division Vice President-Nuclear Stations or his designated alternate. The incident shall be promptly reviewed pursuant to Specification 6.1.G.2.a(5) and a separate report for each reportable occurrence shall be prepared in accordance with the requirements of Specification 6.6.B.

6.4 Action to be Taken in the Event of a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately and reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Division Vice President-Nuclear Stations or his designated alternate. The incident shall be reviewed pursuant to Specification 6.1.G.1.a and 6.1.G.2.a and a separate report for each occurrence shall be prepared in accordance with Specification 6.6.B.

6.5 Plant Operating Records

A. Record and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years.

1. Records of normal plant operation, including power levels and periods of operation at each power level.
2. Record of principal maintenance activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety.
3. Records and reports of reportable and safety limit occurrences.
4. Records and periodic checks, inspection and/or calibrations performed to verify the Surveillance Requirements (See Section 4 of these Specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded.
5. Records of changes made to the equipment or reviews of tests and experiments to comply with 10 CFR 50.59.
6. Records of radioactive shipments.
7. Records of physic tests and other tests pertaining to nuclear safety.
8. Records of changes to operating procedures.
9. Shift Engineers Logs.
10. By-product material inventory records and source leak test results.