Mr. Thomas C. Bordine
Manager, Licensing
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

October 31, 1996

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT RE: REVISION OF ADMINISTRATIVE CONTROLS AND RELATED TECHNICAL SPECIFICATIONS (TAC NO. M94287)

Dear Mr. Bordine:

The Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated December 11, 1995, as supplemented by letters dated January 18, September 3, October 2, October 18, and October 25, 1996.

The amendment modifies TS Section 6.0 to be consistent with the Standard Technical Specifications (STS) for CE Plants, NUREG-1432, by deleting or relocating requirements that are adequately controlled by existing regulatory requirements, by adding requirements, and by editorially restructuring the TS without affecting their technical content. In addition, containment leak rate testing requirements are revised to allow the Type A integrated leak rate test (ILRT) to be scheduled in accordance with Option B of 10 CFR Part 50, Appendix J.

The staff will require additional information to complete review of proposed changes regarding regenerative heat exchanger inspection, containment tendon testing, effluent release limits, and quality assurance requirements. These proposed changes will be addressed under separate correspondence.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by:
Robert G. Schaaf, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-255
Enclosures: 1. Amendment No. 174 to DPR-20
2. Safety Evaluation
cc w/encl: See next page
Distribution: See attached page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 31, 1996

Mr. Thomas C. Bordine
Manager, Licensing
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

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Robert G. Schaaf, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. 174 to DPR-20
2. Safety Evaluation

cc w/encl: See next page

DATED: October 31, 1996

AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-20-PALISADES

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated December 11, 1995, as supplemented by letters dated January 18, September 3, October 2, October 18, and October 25, 1996, complies 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 application, 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;

   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 to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in Licensee's application dated December 11, 1995, as supplemented by letters dated January 18, September 3, October 2, October 18, and October 25, 1996, and reviewed in the Staff's safety evaluation dated October 31, 1996. This license is also hereby amended by changes to the Technical Specifications as indicated in the

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attachment to the license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-20 is hereby amended to read as follows:

> The Technical Specifications contained in Appendix A, as revised through Amendment No. 174, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. CPCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days from the date of issuance. Implementation shall include the relocation of Technical Specification requirements to the appropriate licensee-controlled documents as identified in the Licensee's application dated December 11, 1995, as supplemented by letters dated January 18, September 3, October 2, October 18, and October 25, 1996, and reviewed in the Staff's safety evaluation dated October 31, 1996

FOR THE NUCLEAR REGULATORY COMMISSION

Robert G. Schaaf, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 31, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

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1.0 DEFINITIONS

The following terms are defined for uniform interpretation of these Technical Specifications.

ASSEMBLY RADIAL PEAKING FACTOR - F_r^A

ASSEMBLY RADIAL PEAKING FACTOR shall be the maximum ratio of the power generated in any fuel assembly, to the average fuel assembly power. (Each of these power terms shall be integrated over core height and shall include tilt.)

AVERAGE DISINTEGRATION ENERGY - $\bar{E}$

AVERAGE DISINTEGRATION ENERGY shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

AXIAL OFFSET or AXIAL SHAPE INDEX - AO or ASI

AXIAL OFFSET or AXIAL SHAPE INDEX shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor, alarm, interlock, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated. Neutron detectors may be excluded from CHANNEL CALIBRATIONS.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and status with other indications and status derived from independent instrument channels measuring the same parameter. A CHANNEL CHECK shall include verification that the monitored parameter is within limits imposed by the Technical Specifications.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel to verify that it is OPERABLE, including any alarm and trip initiating function.

COLD SHUTDOWN

The COLD SHUTDOWN condition shall be when the primary coolant is at SHUTDOWN BORON CONCENTRATION and T_{ave} is less than 210°F.

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY is defined to exist when all the following are true:

a. All nonautomatic containment isolation valves and blind flanges are closed (OPERABLE) except as noted in Table 3.6.1.

b. The equipment hatch is properly closed and sealed.

c. At least one door in each personnel air lock is properly closed and sealed.

d. All automatic containment isolation valves are OPERABLE (as demonstrated by satisfying isolation times specified in Table 3.6.1 and leakage criterion in Specification 4.5.2) or are locked closed.

e. The uncontrolled containment leakage satisfies Specification 4.5.

CONTROL RODS

CONTROL RODS shall be all full-length shutdown and regulating rods.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

HOT SHUTDOWN

The HOT SHUTDOWN condition shall be when the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{ave} is greater than 525°F.

HOT STANDBY

The HOT STANDBY condition shall be when T_{ave} is greater than 525°F and any of the CONTROL RODS are withdrawn and the neutron flux power range instrumentation indicates less than 2% of RATED POWER.

LOW POWER PHYSICS TESTING

LOW POWER PHYSICS TESTING shall be testing performed under approved written procedures to determine CONTROL ROD worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of RATED POWER, not including decay heat and PCS T_{ave} and PCS pressure shall be in the range of 371°F to 538°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions in these Technical Specifications.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE, or have OPERABILITY, when it is capable of performing its specified functions, and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified functions are also capable of performing their related support functions.

POWER OPERATION

The POWER OPERATION condition shall be when the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of RATED POWER.

QUADRANT POWER TILT - T_q

QUADRANT POWER TILT shall be the algebraic ratio of quadrant power minus average quadrant power, to average quadrant power.

RATED POWER

RATED POWER shall be a steady state reactor core output of 2530 MW_t.

REACTOR CRITICAL

The reactor is considered critical for purposes of administrative control when the neutron flux wide range channel instrumentation indicates greater than 10^{-4}% of RATED POWER.

REFUELING BORON CONCENTRATION

REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of at least 1720 ppm AND sufficient to assure the reactor is subcritical by ≥ 5% $\Delta\rho$ with all CONTROL RODS withdrawn.

REFUELING OPERATION

A REFUELING OPERATION shall be any operation involving movement of core components (except for incore detectors) when the reactor vessel head is untensioned or removed with fuel in the reactor vessel.

REFUELING SHUTDOWN

The REFUELING SHUTDOWN condition shall be when the primary coolant is at REFUELING BORON CONCENTRATION and T_{ave} is less than 210°F.

SHUTDOWN BORON CONCENTRATION

SHUTDOWN BORON CONCENTRATION shall be a Primary Coolant System boron concentration sufficient to assure the reactor is subcritical by ≥ 2% $\Delta\rho$ with all CONTROL RODS in the core and the highest worth CONTROL ROD fully withdrawn.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming that all CONTROL RODS are fully inserted except for the single highest worth CONTROL ROD which is assumed to be withdrawn.

TOTAL RADIAL PEAKING FACTOR - F_r^T

The TOTAL RADIAL PEAKING FACTOR shall be the maximum product of the ratio of individual assembly power to core average assembly power, times the highest local peaking factor integrated over the total core height, including tilt. Local peaking factor is defined as the maximum ratio of an individual fuel rod power to the assembly average rod power.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Reactor Core

The Minimum DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

Applicability

Safety Limit 2.1 is applicable in HOT STANDBY and POWER OPERATION.

Action

2.1.1 If a Safety Limit is exceeded, the reactor shall be shut down immediately and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).

2.2 Safety Limit - Primary Coolant System Pressure (PCS)

The PCS Pressure shall not exceed 2750 psia.

Applicability

Safety Limit 2.2 is applicable when there is fuel in the reactor.

Action

2.2.1 If a Safety Limit is exceeded, the reactor shall be shut down immediately and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).

2.3 Limiting Safety System Settings - Reactor Protective System (RPS)

The RPS trip setting limits shall be as stated in Table 2.3.1.

Applicability

Limiting Safety System Settings of Table 2.3.1 are applicable when the associated RPS channels are required to be OPERABLE by Specification 3.17.1.

Action

2.3.1 If an RPS instrument setting is not within the allowable settings of Table 2.3.1, immediately declare the instrument inoperable and complete corrective action as directed by Specification 3.17.1.

Amendment No. ~~31~~, ~~25~~, ~~43~~, ~~118~~, ~~137~~, ~~150~~, ~~168~~, 174

Specification

3.17.4 The Accident Monitoring Instruments listed in Table 3.17.4 shall be OPERABLE. (Specifications 3.0.3, 3.0.4, and 4.0.4 do not apply.)

Applicability

Specification 3.17.4 applies when the PCS temperature is > 300°F.

Action

3.17.4.1 With one required channel of functions 1 through 14 inoperable for one or more functions:

a. Restore channel to OPERABLE status within 7 days.

3.17.4.2 With two required channels of functions 1 through 14 inoperable for one or more functions:

a. Restore one channel to OPERABLE status within 48 hours.

3.17.4.3 With position indication inoperable for one or more Containment Isolation Valves:

a. Restore the indication to OPERABLE status or lock the associated valves in the closed position within 7 days.

3.17.4.4 If any action required by 3.17.4.1 through 3.17.4.3 is not met AND the associated completion time has expired,

a. The reactor shall be placed in HOT SHUTDOWN within 12 hours, and

b. The reactor shall be placed in a condition where the affected equipment is not required, within 48 hours.

3.17.4.5 With one channel of functions 16 through 21 inoperable for one or more functions:

a. Restore the channel to OPERABLE status within 7 days.

3.17.4.6 With two required channels of functions 16 through 21 inoperable for one or more functions:

a Restore one channel to OPERABLE status within 48 hours.

3.17.4.7 If any action required by 3.17.4.5 or 3.17.4.6 is not met AND the associated completion time has expired:

a. With two CETs in any one quadrant inoperable, complete Action 3.17.4.4 in lieu of Action 3.17.4.7 c),

b. With two RVWL channels inoperable, initiate alternate monitoring within 48 hours,

c. Submit a report to the NRC in accordance with Specification 6.6.7.

d. Restore the channels to OPERABLE status prior to startup from the next refueling.

Amendment No. ~~136~~, ~~147~~, ~~162~~, 174

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance requirements shall be applicable during the reactor operating conditions associated with individual Limiting Conditions for Operation unless otherwise stated in an individual surveillance requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the operability requirements for a Limiting Condition for Operation. The time limits of the action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The action requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into a reactor operating condition or other specified condition shall not be made unless the Surveillance Requirements associated with a Limiting Condition of Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to plant conditions as required to comply with action requirements.

Amendment No. ~~30~~, ~~51~~, ~~130~~, ~~162~~, ~~171~~, 174

4.0.5 Deleted

Amendment No. ~~130~~, ~~162~~, 174

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance requirements stated in the code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during reactor operating conditions or other conditions for which the requirements of the Limiting Conditions for Operation apply, unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor operating condition or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational condition for which the requirements of the associated Limiting Condition for Operation do not apply, unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the operability requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be operable when Surveillance Requirements have

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into a reactor operating condition or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component operability requirements or parameter limits are met before entry into an operational condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in reactor operating conditions or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with action requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower operational condition.

4.1 OVERPRESSURE PROTECTION SYSTEM TESTS

Surveillance Requirements

In addition to the requirements of The Inservice Inspection and Testing Program, Specification 6.5.7, each PORV flow path shall be demonstrated OPERABLE by:

1. Testing the PORVs in accordance with the inservice inspection requirements for ASME Boiler and Pressure Vessel Code, Section XI, Section IWV, Category B valves.

2. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

3. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.1:

   (a. Performing a complete cycle of the PORV with the plant above COLD SHUTDOWN at least once per 18 months.

   (b. Performing a complete cycle of the block valve prior to heatup from COLD SHUTDOWN, if not cycled within 92 days.

4. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.2:

   (a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days.

   (b. Verifying the associated block valve is open at least once per 72 hours.

5. Both High Pressure Safety Injection pumps shall be verified incapable of injection into the PCS at least once per 12 hours, unless the reactor head is removed, when either PCS cold leg temperature is < 300°F, or when both shutdown cooling suction valves, MO-3015 and MO-3016, are open.

Basis

With the reactor vessel head installed when the PCS cold leg temperature is less than 300°F, or if the shutdown cooling system isolation valves MO-3015 and MO-3016 are open, the start of one HPSI pump could cause the Appendix G or the shutdown cooling system pressure limits to be exceeded; therefore, both pumps are rendered inoperable.

Amendment No. ~~130~~, ~~149~~, ~~160~~, ~~162~~, ~~163~~, ~~164~~, ~~171~~, 174

Table 4.2.3

VENTILATION SYSTEM TESTS

The Control Room Ventilation and Isolation System and the Fuel Storage Area HEPA/Charcoal Exhaust System shall be demonstrated to be OPERABLE by the following tests:

1. Performing required Control Room Ventilation and Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program.

2. At least once per refueling cycle by:

   a. Verifying that on a containment high-pressure and high-radiation test signal, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

   b. Verifying that the Control Room Ventilation system maintains the Control Room at a positive pressure ≥ 1/8 inch WG relative to the outside atmosphere during system emergency mode operation.

   c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.

3. Verifying that the Control Room temperature is ≤ 90°F; once per 12 hours.

Basis - Table 4.2.2 Item 12 - Trisodium Phosphate (TSP) Tests

Item 12.a - TSP quantity verification

Verification of the quantity of TSP in the baskets ensures that neither leakage nor other water sources in the containment reduce the basket content below the required minimum. This requirement ensures that there is an adequate quantity of TSP to adjust the pH of the post LOCA sump solution to a value between 7.0 and 8.0.

Item 12.b - TSP quality verification

Periodic testing is performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. Satisfactory completion of this test assures that the TSP in the baskets is "active" as required by Specification 3.19.

Adequate solubility is verified by submerging a representative sample of TSP from one of the baskets in containment in un-agitated borated water heated to a temperature representing post-LOCA conditions; the TSP must completely dissolve within a 4 hour period. The test time of 4 hours is specified to allow time for the dissolved TSP to naturally diffuse through the un-agitated test solution. Agitation of the test solution during the solubility verification is prohibited, since an adequate standard for the agitation intensity (other than no agitation) cannot be specified. The flow and turbulence in the containment sump during recirculation would significantly decrease the time required for the TSP to dissolve.

Adequate buffering capability is verified by a measured pH of the sample solution, following the solubility verification, between 7 and 8. The sample is cooled and thoroughly mixed prior to measuring pH.

The quantity of the TSP sample, and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions.

4.3 SYSTEMS SURVEILLANCE

APPLICABILITY

Applies to preoperational and inservice structural surveillance of the reactor vessel and other Class 1, Class 2 and Class 3 system components.

OBJECTIVE

To insure the integrity of the Class 1, Class 2 and Class 3 piping systems and components.

SPECIFICATIONS

a,b,c,d,e - Deleted

f. A 100% volumetric examination of the regenerative heat exchanger primary side shell to tube-sheet welds and primary head, shall be performed at least once each 5 years.

g. A surveillance program to monitor radiation induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained as described in Section 4.5.3 of the FSAR.

h. Periodic leakage testing[a],[b] on each check valve listed in Table 4.3.1 shall be accomplished prior to returning to the Power Operation Condition after every time the plant has been placed in the Refueling Shutdown Condition, or the Cold Shutdown Condition for more than 72 hours if such testing has not been accomplished within the previous 9 months, and prior to returning the check valves to service after maintenance, repair or replacement work is performed on the valves.

i. Whenever integrity of a pressure isolation valve listed in Table 4.3.1 cannot be demonstrated and credit is being taken for compliance with Specification 3.3.3.b, the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily and the position of the other closed valve located in that pressure line shall be recorded daily.

j. Following each use of the LPSI system for shutdown cooling, the reactor shall not be made critical until the LPSI check valves (CK-3103, CK-3118, CK-3133 and CK-3148) have been verified closed.

[a]To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

[b]Reduced pressure testing is acceptable (see footnote 5, Table 4.3.1). Minimum test differential pressure shall not be less than 150 psid.

Basis

The inspection program specified places major emphasis on the areas of highest stress concentration as determined by general design evaluation and experience with similar systems.[1] In addition, that portion of the reactor vessel shell welds which will be subjected to a fast neutron dose sufficient to change ductility properties will be inspected. The inspections will rely primarily on ultrasonic methods utilizing up-to-date analyzing equipment and trained personnel. To the extent applicable, based upon the existing design and construction of the plant, the requirements of Section XI of the Code shall be complied with. Significant exceptions are detailed in the requests for relief which have received NRC approval and are contained in the Class 1, Class 2 and Class 3 Long-Term Inspection Plans.

Valve Testing

To ensure the continued integrity of selected check valves which are relied upon to preclude a potential LOCA outside containment, special requirements for periodic leak tests are specified. In addition a valve disk position check for the LPSI check valves is specified following each use of the LPSI system for shutdown cooling. This position check ensures that the four LPSI check valves have reclosed upon cessation of shutdown cooling flow.

References

(1) FSAR, Section 4.5.6

(2) Deleted

(3) Systematic Evaluation Program Topic V-II.A, NRC letter to the licensee transmitting the final topic evaluation dated November 9, 1981.

TABLE 4.3.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum [a] Allowable Leakage
High Pressure Safety Injection		
Loop 1A, Cold Leg	3101	5.0 gpm
	3104	5.0 gpm
Loop 1B, Cold Leg	3116	5.0 gpm
	3119	5.0 gpm
Loop 2A, Cold Leg	3131	5.0 gpm
	3134	5.0 gpm
Loop 2B, Cold Leg	3146	5.0 gpm
	3149	5.0 gpm
Low Pressure Safety Injection		
Loop 1A, Cold Leg	3103	5.0gpm
Loop 1B, Cold Leg	3118	5.0gpm
Loop 2A, Cold Leg	3133	5.0gpm
Loop 2B, Cold Leg	3148	5.0gpm

Footnote:

[a]

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.
5. Measured leakage rates must be adjusted for test pressures less than the maximum potential pressure differential across the valve by assuming leakage to be directly proportional to the pressure differential to the one-half power.

4.4 Deleted

4.5 CONTAINMENT TESTS

4.5.1 Integrated Leakage Rate Tests

The containment integrated leak rate testing shall be performed in accordance with the Containment Leak Rate Testing Program.

4.5.2 Local Leak Detection Tests

a. Test

(1) Local leak rate tests shall be performed at a pressure of not less than 55 psig.

(2) Local leak rate tests for checking air lock door seals within 72 hours of each door opening shall be performed at a pressure of not less than 10 psig.

(3) Acceptable methods of testing are halogen gas detection, soap bubble, pressure decay, or equivalent.

(4) The local leak rate shall be measured for each of the following components:

(a) Containment penetrations that employ resilient seal gaskets, sealant compounds, or bellows.

(b) Air lock and equipment door seals.

(c) Fuel transfer tube.

(d) Isolation valves on the testable fluid systems' lines penetrating the containment.

(e) Other containment components which require leak repair in order to meet the acceptance criterion for any integrated leak rate test.

b. Acceptance Criteria

(1) The total leakage from all penetrations and isolation valves shall not exceed 0.60 L_a.

(2) The leakage for an air lock door seal test shall not exceed 0.023 L_a.

4.5.2 Local Leak Detection Tests (continued)

c. Corrective Action

(1) If at any time it is determined that 0.60 L_a is exceeded, repairs shall be initiated immediately. If repairs are not completed and conformance to the acceptance criterion of 4.5.2.b(1) is not demonstrated with 48 hours, the Plant shall be placed in at least hot shutdown within the next 6 hours and in at least cold shutdown within the following 30 hours.

(2) If at any time it is determined that total containment leakage exceeds L_a, within one hour action shall be initiated to bring the Plant to hot shutdown within the next six (6) hours and cold shutdown within the following thirty (30) hours.

(3) If air lock door seal leakage is greater than 0.023 L_a, repairs shall be initiated immediately to restore the door to less than specification 4.5.2.b(2). In the event repairs cannot be completed within 7 days, the Plant shall be brought to a hot shutdown condition within the next six (6) hours and cold shutdown within the following thirty (30) hours.

If air lock door seal leakage results in one (1) door causing total containment leakage to exceed 0.60 L_a, the door shall be declared inoperable and the remaining operable door shall be immediately locked closed and tested within four (4) hours. As long as the remaining door is found to be operable, the provisions of 4.5.2.c(2) do not apply. Repairs shall be initiated immediately to establish conformance with specification 4.5.2.b(1). In the event conformance to this specification cannot be established within 48 hours the Plant shall be brought to a hot shutdown within the next 6 hours and cold shutdown within the following 30 hours.

4.5.2 Local Leak Detection Tests (continued)

d. Test Frequency

(1) Individual penetrations and containment isolation valves shall be leak rate tested at a frequency of at least every six months prior to the first postoperational integrated leak rate test and at a frequency of at least every refueling thereafter, not exceeding a two-year interval, except as specified in (a) and (b) below:

(a) The containment equipment hatch and the fuel transfer tube shall be tested at each refueling shutdown or after each time used, if that be sooner.

(b) A full air lock penetration test shall be performed at six-month intervals. During the period between the six-month tests when containment integrity is required, a reduced pressure test for the door seals or a full air lock penetration test shall be performed within 72 hours after either each air lock door opening or the first of a series of openings.

(2) Each three months the isolation valves must be stroked to the position required to fulfill their safety function unless it is established that such operation is not practical during plant operation. The latter valves shall be full-stroked during each cold shutdown.

4.5.3 Containment Isolation Valves

a. The isolation valves shall be demonstrated operable by performance of a cycling test and verification of isolation time for auto isolation valves prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit.

b. Each isolation valve shall be demonstrated operable by verifying that on each containment isolation right channel or left channel test signal, applicable isolation valves actuate to their required position during cold shutdown or at least once per refueling cycle.

c. The isolation time of each power operated or automatic valve shall be determined to be within its limit as specified in Table 3.6.1 when tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

4.5.4 Surveillance for Prestressing System

a. Tendon inspection shall be accomplished at five-year intervals for the life of the plant. The scheduled inspection dates for all subsequent inspections may be varied by not more than plus or minus one year from the base schedule.

b. The surveillance tendons shall be randomly but representatively selected from each of the following groups:

1. A minimum of 4 dome tendons including one from each dome tendon group.
2. A minimum of 4 vertical tendons.
3. A minimum of 5 hoop tendons.

For each inspection, the tendons shall be selected on a random basis except that those tendons whose routing has been modified to clear penetrations shall be excluded from the sample.

c. During each tendon inspection, the following field testing shall be performed:

1. Lift-off readings shall be taken for each of the surveillance tendons. The tests shall include the following actions:

(a) One tendon, randomly selected from each group of tendons during each inspection, shall be subjected to essentially complete detensioning to identify broken or damaged wires.

(b) The simultaneous measurement of elongation and jacking force during retensioning shall be made at a minimum of three approximately equally spaced levels of force between the seating force and zero.

2. While the tendon is in the detensioned state, each wire in the tendon will be checked for continuity.

3. Three wires, one from each of a vertical, a hoop and a dome tendon will be removed and identified for inspection. At each successive surveillance, the wires will be selected from different tendons. Each of the inspection wires removed will be visually inspected for evidence of corrosion or other deleterious effects and samples taken for laboratory testing.

4. The sheathing filler shall be inspected visually for color and coverage and samples shall be obtained for laboratory testing.

5. Tendon anchorage hardware such as bearing plates, stressing washers, shims and buttonheads shall be visually inspected for evidence of corrosion or other deleterious effects.

4.5.4 Surveillance for Prestressing System (continued)

d. Following the field testing of 4.5.4c, the following laboratory testing shall be done:

1. Three tensile test specimens shall be cut from each of the three inspection wires removed (one from each end and one from the middle). One additional specimen shall be cut from the wire determined by field visual inspection to have the greatest amount of corrosion. Each of the wire samples shall be tested for ultimate strength, yield strength, and elongation.

2. The sheathing filler samples shall be taken from each end of each tendon examined. Vertical tendon samples shall be taken from the lower end. Samples shall be thoroughly mixed and analyzed for reserve alkalinity, water content, and concentration of water soluble chlorides, nitrates, and sulfides. Analyses shall be performed in accordance with the procedures and within the acceptance limits specified in ASME Code Section XI, Table IWL-2525-1.

   Procedures shall be established to minimize voids and to assure that the volume of sheathing filler removed has been replaced upon completion of the inspection and amounts documented.

e. Acceptance criteria shall be as follows:

1. The average of all measured tendon forces for each type of tendon shall be equal to or greater than the minimum required prestress level, of 584 kips per tendon for dome tendons and, 615 kips per tendon for hoop and vertical tendons. The measured force in each individual tendon shall not be less than 95% of the predicted force, or

   (a) the measured force in not more than one tendon is between 90% and 95% of the predicted force, and

   (b) The measured forces in two tendons located adjacent to the tendon in (a) above are not less than 95% of the predicted forces, and

   (c) the measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

   If measured force in any tendon is less than 90% of its predicted force, the tendon shall be completely detensioned and a determination shall be made as to the cause of such an occurrence and corrective action shall be taken. In addition, all such tendons shall have their forces measured as additional tendons in the next scheduled inspection period. The Commission shall be notified in accordance with Paragraph 4.5.4f.

4.5.4 Surveillance for Prestressing System (continued)

2. Inspection wires shall indicate no significant loss of section by corrosion or pitting.

3. Tensile test specimens cut from inspection wires shall be tested for ultimate strength. Failure at less than 11.78 kips of any one of the test samples requires the Commission be notified in accordance with specification 4.5.4f.

4. Tendon anchorage hardware shall be free of significant corrosion, pitting, cracks or other deleterious effects.

f. If any element of the prestressing system fails to meet the acceptance criteria of 4.5.4e., the reporting provisions of 10 CFR 50.73 shall apply.

4.5.5 End Anchorage Concrete Surveillance

a. A VT-1 visual examination shall be performed on the end anchorage concrete surface at the surveillance tendon anchor points for signs of cracking, popouts, spalling, or corrosion. Concrete cracks having widths greater than 0.010 shall be evaluated and documented.

b. The end anchorage concrete surveillance inspection interval shall be the same as tendon surveillance interval.

c. Acceptance criteria

1. Crack widths shall be measured by using' optical comparators or wire feeler gauge. Movements shall be measured by using demountable mechanical extensometers.

2. Concrete anchorage areas are acceptable if no concrete cracks are wider than 0.010 inches and no signs of new or progressive deterioration since the previous inspection are found.

3. Concrete surface conditions exceeding those stated in 4.5.5c.2 above shall be evaluated for the effect on tendon and containment structural integrity. The results of evaluation shall be included in the final surveillance report.

4.5.6 Dome Delamination Surveillance

If, as a result of a prestressing system inspection under Section 4.5.4, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits of Specification 4.5.4, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC.

Basis

The containment is designed for an accident pressure of 55 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 104°F. With these initial conditions, following a LOCA, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is 283°F.

Prior to initial operation, the containment was strength-tested at 63 psig and then leak rate tested. The design objective of this preoperational leak rate test was established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which were independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours. With this leakage rate and with a reactor power level of 2530 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the Maximum Hypothetical Accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leak rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by CHP only. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.[4]

The Type A test requirements including pretest test methods, test pressure, acceptance criteria, and reporting requirements are in accordance with the Containment Leak Rate Testing Program.[5,6]

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The specified frequency is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which in consistent with 0.1% leakage at design basis accident (DBA) conditions: and (c) absence of any significant stresses in the liner during reactor operation.

Basis (continued)

Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value ($0.60L_a$) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the Containment Structural Integrity Surveillance Program which provides assurance that an important part, of the structural integrity of the containment is maintained.

The basis for specification of a total leakage rate of 0.60 L_a from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests.

The basis for specification of an airlock door seal leakage rate of 0.023 L_a is to provide assurance that the failure of a single airlock door will not result in the total containment leakage exceeding 0.6 L_a. The seven (7) day LCO specified for exceeding the airlock door leakage limit is acceptable since it requires that the total containment leakage limit is not exceeded.

A reduction in prestressing force and change in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. The inspection results for each tendon inspected shall be recorded on the forms provided for that purpose and comparison will be made with previous test results and the initial quality control records.

Force-time records will be established and maintained for each of the tendon groups, dome, hoop and vertical. If the force measured for a tendon is less than the lower bound curve of the force-time graph, two adjacent tendons will be tested. If either of the adjacent or more than one of the original sample population falls below the lower bound of the force-time graph, an investigation will be conducted before the next scheduled surveillance. The investigation shall be made to determine whether the rate of force reduction is indeed occurring for other tendons. If the rate of reduction is confirmed, the investigation shall be extended so as to identify the cause of the rate of force reduction. The extension of the investigation shall determine the needed changes in the surveillance inspection schedule and the criteria and initial planning for corrective action.

If the force measured for a tendon at any time exceeds the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

If the comparison of corrosion conditions, including chemical tests of the corrosion protection material, indicate a larger than expected change in the conditions from the time of installation or last surveillance inspection, and investigation shall be made to detect and correct the causes.

Basis (continued)

The prestressing system is a necessary strength element of the plant safeguards and it is considered desirable to confirm that the allowances are not being exceeded. The technique chosen for surveillance is based upon the rate of change of force and physical conditions so that the surveillance can either confirm that the allowances are sufficient, or require maintenance before minimum levels of force or physical conditions are reached.

The end anchorage concrete is needed to maintain the prestressing forces. The design investigations concluded that the design is adequate. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. At the time of initial pressure testing, the containment building had been subjected to temperature gradients equivalent to those for normal operating conditions while the prestressing tendon loads are at their maximum.

However, after the initial pressure test both concrete creep and prestressing losses increase with the greatest rapidity and result in a redistribution of the stresses and a reduction in end anchor force. Because of the importance of the containment and the fact that the design was new, it was considered prudent to continue the surveillance after the initial period.

Containment dome delamination inspections performed in 1970 and 1982 have confirmed that no concrete delamination has occurred. The possibility that delamination might occur in the future is remote because dome tendon prestress forces gradually diminish through normal tendon relaxation and concrete strength normally increases over time. To account for this remote possibility, however, an additional delamination inspection will be performed in the event that 5% or more of the installed tendons must be retensioned to compensate for excessive loss of prestress. This inspection would be to confirm that any systematic excessive prestress loss did not result from delamination and that the retensioning process did not result in delamination.

References

(1) Updated FSAR Section 5.8.2.

(2) Updated FSAR Section 5.8.8

(3) Updated FSAR 14.22

(4) Updated FSAR Section 8.5.1.2

(5) 10 CFR Part 50, Appendix J.

(6) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", September 1995.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Surveillance Requirements

4.6.1 Safety Injection System

a. System tests shall be performed at each reactor refueling interval. A test safety injection signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this test. The system will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing (ie, the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

4.6.2 Containment Spray System

a. System test shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every ten years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.6.3 Pumps

a. The safety injection pumps, shutdown cooling pumps, and containment spray pumps shall be started at intervals not to exceed three months. Alternate manual starting between control room console and the local breaker shall be practiced in the test program.

b. Acceptable levels of performance shall be that the pumps start, reach their rated heads on recirculation flow, and operate for at least fifteen minutes.

4.6.4 Valves

a. Each Safety Injection Tank flow path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying each motor operated isolation valve is open by observing valve position indication and valve itself, and locking open the associated circuit breakers.

b. The Low Pressure Safety Injection flow path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying flow control valve CV-3006 is open, and its air supply is isolated.

Surveillance Requirements (continued)

4.6.4 Valves (continued)

c. The safety injection recirculation path shall be verified OPERABLE within 7 days prior to each reactor startup by verifying valves CV-3027 and 3056 are open and their switches HS-3027A, HS-3027B, HS-3056A, and HS-3056B are open.

d. Each Containment Spray Valve manual control shall be verified to be OPERABLE at least once each refueling by cycling each valve from the control room while observing valve operation at least each 18 months.

4.6.5 Containment Air Cooling System

a. Emergency mode automatic valve and fan operation will be checked for OPERABILITY during each refueling shutdown.

b. Each fan and valve required to function during accident conditions will be exercised at intervals not to exceed three months.

4.6 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS TESTS

Basis

The safety injection system and the containment spray system are principal plant safety features that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a containment spray system test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore, to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The refueling interval systems tests demonstrate proper automatic operation of the safety injection and containment spray systems. A test signal is applied to initiate automatic action and verification made that the components receive the Safety Injection Signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1,2]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked each shift and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment that more frequent testing would not significantly increase the reliability (ie, the probability that the component would operate when required), yet more frequent test would result in increased wear over a long period of time. Verification that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at least every ten years thereafter. Since the material is all stainless steel, normally in a dry condition, and with no plugging mechanism available, the retest every ten years is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the service water system and the containment air coolers. The SI tanks are a passive safety feature. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR, Section 6.1.3.
(2) FSAR, Section 6.2.3.

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4.14 STEAM GENERATOR SURVEILLANCE

4.14.1 Verify Steam Generator tube integrity is acceptable in accordance with the Inservice Inspection and Testing Program, Specification 6.5.7, and the Steam Generator Tube Surveillance Program, Specification 6.5.8.

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4.16 INSERVICE INSPECTION PROGRAM FOR SHOCK SUPPRESSORS (Snubbers)

Applicability

Applies to periodic surveillance of safety-related snubbers as described per Specification 3.20.

4.16.1 Specifications

Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 6.5.7. As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

a. Visual Inspection

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the following paragraph:

If one or more unacceptable snubbers are found, the next inspection interval shall be 2/3 (-25%) of the previous interval. If no unacceptable snubbers are found, the next interval may be doubled (-25%), but not to exceed 48 months. The interval extension provisions of Technical Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

Inspections performed before the interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections, performed before the original required time interval has elapsed (nominal time less 25%), may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

b. Visual Inspection Acceptance Criteria

Visual inspection shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Technical Specification 4.16.1d or 4.16.1e, as applicable. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval.

4.16 INSERVICE INSPECTION PROGRAM FOR SHOCK SUPPRESSORS (snubbers)

4.16.1 f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each safety related snubber in use in the plant shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled 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 re-evaluation, replacement or reconditioning shall be indicated in the records.

6.1 RESPONSIBILITY

6.1.1 The plant superintendent shall be responsible for overall plant operation and shall delegate in writing the succession for this responsibility during his absence.

The plant superintendent or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

6.1.2 The Shift Supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the plant is above COLD SHUTDOWN, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the plant is in COLD SHUTDOWN, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the Palisades plant.

a. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented, and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key positions, or in equivalent forms of documentation. These requirements and the plant specific equivalent of those titles referred to in these Technical Specifications shall be documented in the FSAR.

b. The plant superintendent shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out radiation safety and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating above COLD SHUTDOWN.

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is above COLD SHUTDOWN, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i), and 6.2.2.a and 6.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the requirements.

d. A radiation safety technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions (e.g., licensed SROs, licensed ROs, radiation safety personnel, auxiliary operators, and key maintenance personnel).

In the event that overtime is used, the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

6.2.2.e Plant Staff (Continued)

Any deviations from the overtime guidelines shall be authorized in advance by the plant superintendent or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the plant superintendent or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The operations manager or an assistant operations manager shall hold an SRO license. The individual holding the SRO license shall be responsible for directing the activities of the licensed operators.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.3.2 The radiation safety manager shall meet the qualifications of a Radiation Protection Manager as defined in Regulatory Guide 1.8, September 1975. For the purpose of this section, "Equivalent," as utilized in Regulatory Guide 1.8 for the bachelor's degree requirement, may be met with four years of any one or combination of the following: (a) Formal schooling in science or engineering, or (b) operational or technical experience and training in nuclear power.

6.3.3 The Shift Technical Advisor shall have a bachelor's degree or equivalent and the Shift Engineer shall have a bachelor's degree in a scientific or engineering discipline. Specific training for both the Shift Technical Advisor and the Shift Engineer shall include plant design, operations, and response and analysis of the plant for transients and accidents. The Shift Engineer shall hold a Senior Reactor Operator license.

6.3.4 The plant staff who perform reviews which ensure compliance with 10 CFR 50.59 shall meet or exceed the minimum qualifications of ANS 3.1-1987, Section 4.7.1 and 4.7.2. A Senior Reactor Operator license or certification shall be considered equivalent to a bachelors degree for the purpose of this specification.

6.4 PROCEDURES

6.4.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Site Fire Protection Program implementation.

e. All programs specified in Specification 6.5.

f. Site Security Plan implementation.

g. Site Emergency Plan implementation.

6.4.2 Procedures and changes shall be approved prior to implementation by the appropriate* senior department manager predesignated by the Plant General Manager subject to the reviews per Specifications 6.8.1.6 and 6.8.3.

6.4.3 Temporary changes to procedures of Specification 6.4.1 above may be made provided:

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

b. The change is approved by two members (or designated alternates) of the PRC, at least one of whom holds a Senior Reactor Operator License.

c. The change is documented, subsequently reviewed by Plant Safety and Licensing within 30 days of issuance and approved by the appropriate* senior department manager predesignated by the Plant General Manager.

* The determination of the appropriate senior department manager is based on the activities addressed by the specific procedure and will be predesignated in writing by the Plant General Manager.

6.5 PROGRAMS AND MANUALS

The following programs shall be established, implemented, and maintained:

6.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain (1) the radioactive effluent controls and radiological environmental monitoring activities and (2) descriptions of the information that should be included in the Radiological Environmental Operating Report, and Radioactive Effluent Release Report required by Specification 6.6.2. and Specification 6.6.3.

c. Changes to ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the changes, and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after approval by the plant superintendent.

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray System, the Safety Injection System, the Shutdown Cooling System, and the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

6.5.3 Post Accident Sampling Program

This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas and which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analytic equipment.

6.5.4 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the Offsite Dose Calculation Manual (ODCM), (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 20, Appendix B, TableII, Column 2.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106, and with the methodology and parameters in the ODCM,

d. Limitation on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas conforming to 10 CFR 50, Appendix I,

e. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the doses associated with 10 CFR 20, Appendix B, Table II, Column 1.

f. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

g. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

h. Limitations on the annual doses or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

6.5.5 Reserved

6.5.6 Primary Coolant Pump Flywheel Surveillance Program

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each refueling .

6.5.7 Inservice Inspection and Testing Program

This program provides controls for inservice inspection and testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda (B&PV Code) as follows:

B&PV Code terminology for inservice testing activities	Required interval for performing inservice testing activities
Weekly	≤ 7 days
Monthly	≤ 31 days
Quarterly or every 3 months	≤ 92 days
Semiannually or every 6 months	≤ 184 days
Every 9 months	≤ 276 days
Yearly or annually	≤ 366 days
Biennially or every 2 years	≤ 731 days

b. The provisions of Surveillance Requirement 4.0.2 are applicable to the above required intervals for performing inservice testing activities;

c. The provisions of Surveillance Requirement 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the B&PV Code shall be construed to supersede the requirements of any Technical Specification.

6.5.8 Steam Generator Tube Surveillance Program

This program provides controls for surveillance testing of the Steam Generator (SG) tubes to ensure that the structural integrity of this portion of the Primary Coolant System (PCS) is maintained. The program shall contain controls to ensure:

a. Steam Generator Tube Sample Selection and Inspection

The inservice inspection may be limited to one SG on a rotating schedule encompassing 6% of the tubes if the results of previous inspections indicate that both SGs are performing in a like manner. If the operating conditions in one SG are found to be more severe than those in the other SG, the sample sequence shall be modified to inspect the most severe conditions.

The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 6.5.8-1. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

2. The first sample of tubes selected for each inservice inspection of each SG shall include:

   a) All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

   b) Tubes in those areas where experience has indicated potential problems.

   c) A tube inspection shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

6.5.8 Steam Generator Tube Surveillance Program (continued)

3. The tubes selected as the second and third samples (if required by Table 6.5.8-1) during each inservice inspection may be subjected to a partial tube inspection provided:

   a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   b) The inspections include those portions of the tubes where imperfections were previously found.

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

b. Inspection Frequencies

The above required inservice inspection of SG tubes shall be performed at the following frequencies:

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspections results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

6.5.8 Steam Generator Tube Surveillance Program (continued)

2. If the results of the inservice inspection of a SG conducted in accordance with Table 6.5.8-1 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 6.5.8.b.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 6.5.8-1 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.5.

   b) A seismic occurrence greater than the Operating Basis Earthquake.

   c) A loss-of-coolant accident resulting in initiation of flow of the engineered safeguards.

   d) A main steam line or main feedwater line break.

c. Acceptance Criteria

1. As used in this Specification:

   a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

   b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

   c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

   d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6.5.8 Steam Generator Tube Surveillance Program (continued)

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

g) Unserviceable described the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 6.5.8.b.3, above.

h) Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

i) Preservice Inspection means an inspection of the full length of each tube in SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the shop hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 6.5.8-1.

TABLE 6.5.8-1
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first Sample
			C-3	Perform action for C-3 result of first Sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. 24 hour verbal notification to NRC with written follow up within next 30 days	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes each S.G. and plug defective tubes.	N/A	N/A

S = 6/n % Where n is the number of steam generators inspected during an inspection

6.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

6.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Pool Ventilation (FPV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the FPV when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
V-8A or V-8B	7300 ± 20%
V-8A and V-8B	10,000 ± 20%
V-95 or V-96	12,500 ± 10%

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the FPV when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
V-8A and V-8B	10,000 ± 20%
V-26A and V-26B	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
VF-66	6.00%	95%
VFC-26A and VFC-26B	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_2O)	Flowrate (CFM)
V-8A and V-8B	6.0	10,000 ± 20%
VF-26A and VF-26B	8.0	3200 +10% -5%

e. Demonstrate that the heaters for each of the ventilation systems dissipate the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
VHX-26A and VHX-26B	15 kW

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

6.5.11 Reserved

6.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.5.12.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.13 Reserved

6.5.14 Containment Leak Rate Testing Program

Programs shall be established to implement the leak rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The Type A test program shall meet the requirements of 10 CFR 50, Appendix J, Option B and shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program, dated September 1995." The Type B and Type C test program shall meet the requirements of 10 CFR 50, Appendix J, Option A, as modified by the exemption from certain requirements of 10 CFR 50 Appendix J which was granted in an NRC letter to Consumers Power Company dated December 6, 1989.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.64 psig (FSAR Table 14.18.1-4).

The maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leak rate acceptance criteria are:

a. Containment leak rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock leak rate acceptance criteria is $\leq 0.023\ L_a$ for each door, when pressurized to ≥ 10 psig.

The Surveillance interval extensions of LCO 4.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

The provisions of LCO 4.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

6.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program, CPC-2A. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.6.1 Occupational Radiation Exposure Report

This report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignment to various duty functions may be estimates based on pocket dosimeter, electronic 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. The report shall be submitted by April 30 of each year.

6.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

6.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the NRC to arrive no later than the fifteenth of each month following the calendar month covered by the report.

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1 ASI Limits.
3.10.5 Regulating Group Insertion Limits
3.23.1 Linear Heat Rate (LHR) Limits
3.23.2 Radial Peaking Factor Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs 3.1.1, 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.1.1, 3.23.1, & 3.23.2)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs 3.10.5, 3.23.1, & 3.23.2)

a) XN-NF-82-20(A), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

b) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

c) XN-NF-81-58(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

6.6.5 COLR (continued)

d) XN-NF-85-16(A), "PWR 17x17 Fuel Cooling Tests Program," Volume 1 and Supplements 1(P)(A), 2(P)(A), and 3(P)(A), and Volume 2 and Supplement 1(P)(A); Exxon Nuclear Company.

e) XN-NF-85-105(A), "Scaling of FCTF Based Reflood Heat Transfer Correlation for other Bundle Designs," and Supplement 1(P)(A); Exxon Nuclear Company.

7. XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.10.5, 3.23.1, & 3.23.2)

8. ANF-1224(P)(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)

9. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

10. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

6.6.6 Reserved

6.6.7 Accident Monitoring Instrument Report

When a report is required by Condition 3.17.4.7c, "Accident Monitoring Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

6.6.8 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Liner and Penetration tests within 90 days after completion of the tests.

6.6.9 Steam Generator Tube Surveillance Report

The following reports shall be submitted to the Commission following each inservice inspection of steam generator tubes:

a. The number of tubes plugged in each steam generator shall be reported to the Commission within 15 days following the completion of each inspection, and

b. The complete results of the steam generator tube inservice inspection shall be reported to the Commission within 12 months following completion of the inspection. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 shall require 24 hour verbal notification to the NRC prior to resumption of plant operation. A written followup within the next 30 days shall provide a description of investigations and corrective measures taken to prevent recurrence.

6.7 HIGH RADIATION AREA

6.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Work Request.

6.7.2 In addition to the requirements of Specification 6.7.1, except as allowed by 6.7.3, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

6.7.3 For individual high radiation areas with radiation levels of ≥ 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

6.8 REVIEW AND AUDIT

6.8.1 PLANT REVIEW COMMITTEE (PRC)

6.8.1.1 FUNCTION

The Plant Review Committee (PRC) shall function to advise the Plant General Manager on all matters related to nuclear safety.

6.8.1.2 COMPOSITION

The PRC is composed of nine regular members. The qualification level for PRC members shall be at least equivalent to those described in Section 4.4 of ANSI N18.1-1971. The PRC shall include representatives from the Operations, Radiological Services, Maintenance and Engineering Departments. The Chairman, Alternate Chairmen, and members shall be designated in administrative procedures by the Plant General Manager.

6.8.1.3 ALTERNATES

Alternate members of the PRC shall be appointed in writing by the PRC Chairman to serve on a temporary basis. No more than two alternates shall participate as voting members at any one time in PRC activities.

6.8.1.4 MEETING FREQUENCY

The PRC shall meet at least once per calendar month with special meetings as required.

6.8.1.5 QUORUM

A quorum of the PRC shall consist of the Chairman or alternate and four members or alternates.

6.8.1.6 RESPONSIBILITIES

The PRC shall be responsible for nuclear safety review of:

a. All procedures and programs specified by Specification 6.4 and 6.5 and changes thereto, and any other procedures or changes thereto as determined by the Plant General Manager to affect nuclear safety; all proposed tests or experiments that affect nuclear safety; all proposed changes or modifications to plant systems or equipment that affect nuclear safety; and the Site Emergency Plan.

b. All proposed changes to Operating License and Technical Specifications.

c. Results of investigations of all violations of the Technical Specifications. (A report shall be prepared covering evaluation and recommendations to prevent recurrence and be forwarded to the Vice President - NOD and to the Director, Nuclear Performance Assessment Department (NPAD).)

d. Plant operations to detect potential safety hazards.

e. Reports of special reviews and investigations as requested by the Plant General Manager or NPAD.

f. All reportable events as defined in 10 CFR 50.73.

g. All items identified under Specification 6.8.3.4 as significant to nuclear safety.

h. Monthly reports from Plant Safety and Licensing.

i. Nuclear industry operating experience.

PRC review of the above items may be performed by routing, subject to the requirements of Specification 6.8.1.7. PRC may delegate review of item a. to Plant Safety and Licensing as described in Specification 6.8.3.

6.8.1.7 AUTHORITY

The PRC shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Specifications 6.8.1.6.a. through i. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.8.1.6.a, b, c and g above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President - Nuclear Operations and to the Nuclear Performance Assessment Department of any disagreements between the PRC and the Plant General Manager; however, the Plant General Manager shall have responsibility for the resolution of such disagreements pursuant to Specification 6.1.1 above.

The PRC Chairman may recommend to the Plant General Manager approval of those items identified in Specification 6.8.1.6 above based on a routing review provided the following conditions are met: (1) at least five PRC members including the Chairman and no more than 2 alternates, shall review the item, concur with determination as to whether or not the item constitutes an unreviewed safety question, and provide written comments on the item; (2) all comments shall be resolved to the satisfaction of the reviewers providing the comments; and (3) if the PRC Chairman determines that the comments are significant, the item (including comments and resolutions) shall be recirculated to all reviewers for additional comments.

The item shall be reviewed at a PRC meeting in the event that: (1) Comments are not resolved; or (2) the Plant General Manager overrides the recommendations of the PRC; or (3) a proposed change to the Technical Specifications involves a safety limit, a limiting safety system setting or a limiting condition for operation; or (4) the item was reportable to the NRC.

6.8.1.8 RECORDS

The PRC shall maintain written minutes of each PRC meeting and shall provide copies to the NPAD.

6.8.2 NUCLEAR PERFORMANCE ASSESSMENT DEPARTMENT (NPAD)

6.8.2.1 FUNCTION

The Nuclear Performance Assessment Department (NPAD) shall function to provide independent review of activities in the areas of:

a. Nuclear power plant operation
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Nondestructive testing
f. Instrumentation and control
g. Radiological safety
h. Mechanical and electrical engineering
i. Administrative controls and quality assurance practices
j. Emergency Planning
k. Training

6.8.2.2 COMPOSITION

The NPAD shall include the Director, who reports to the Vice President - NOD, and a full-time staff of Nuclear Performance Specialists reporting to the Director. The Director and the Nuclear Performance Specialists shall meet or exceed the qualifications described in Section 4.7 of ANSI/ANS 3.1-1987. The NPAD shall have no direct responsibility for activities subject to its review.

6.8.2.3 CONSULTANTS

If sufficient expertise is not available within NPAD to review particular issues, the NPAD shall have the authority to utilize consultants or other qualified organizations for expert advice.

6.8.2.4 RESPONSIBILITIES

6.8.2.4.1 REVIEW

The NPAD shall review:

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

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

6.8.2.4.1 REVIEW (Continued)

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

d. Proposed changes to Technical Specifications or the Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affects nuclear safety.

g. All reportable events having nuclear safety significance.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meeting minutes of the Plant Review Committee.

j. Fire Protection Program and Implementing Procedure Changes.

6.8.2.4.2 AUDITS

Audits of operational nuclear safety-related facility activities shall be performed by the NPAD staff under the cognizance of the Nuclear Performance Specialists. These audits shall encompass:

a. The conformance of plant operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The performance of activities required by the Quality Assurance Program Description for Operational Nuclear Power Plants (CPC-2A) to meet the criteria of 10 CFR 50, Appendix B at least once per 24 months.

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

e. The Site Security Plan and implementing procedures (as required by the Site Security Plan) at least once per 12 months.

6.8.2.4.2 AUDITS (Continued)

f. Any other area of plant operation considered appropriate by NPAD or the Vice President - Nuclear Operations.

g. The plant Fire Protection Program and implementing procedures at least once per 24 months.

h. An independent fire protection and loss prevention inspection and audit to be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program to be performed by an outside qualified fire consultant at intervals no greater than 3 years.

j. Radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

Audit reports encompassed by Specification 6.8.2.4.2 above shall be forwarded to the Director NPAD, and Management positions responsible for the areas audited within thirty (30) days after completion of the audit.

6.8.2.4.3 NPAD review of the subjects in Specifications 6.8.2.4.1 and 6.8.2.4.2 shall be performed by an assigned Nuclear Performance Specialist selected on the basis of his technical expertise relative to the subject being reviewed. If the assigned Nuclear Performance Specialist determines the need for interdisciplinary review, a committee consisting of the Director, NPAD, or his designate, and at least four Nuclear Performance Specialists, shall be assigned. Such committee shall meet as conditions requiring interdisciplinary review arise, but no less than twice yearly.

6.8.2.5 AUTHORITY

The NPAD shall report to and advise the Vice President - NOD of significant findings associated with those areas of responsibility specified in Sections 6.8.2.4.1 and Section 6.8.2.4.2.

6.8.2.6 RECORDS

Records of NPAD activities shall be maintained. Reports shall be prepared and distributed as indicated below:

a. The results of reviews, performed pursuant to Section 6.8.2.4.1 and Section 6.8.2.4.2, shall be reported to the Vice President - NOD at least monthly.

b. A report assessing the overall nuclear safety performance of Palisades shall be provided to senior Consumers Power Company management annually.

6.8.3 PLANT SAFETY AND LICENSING

6.8.3.1 FUNCTION

The Plant Safety and Licensing staff shall review proposed changes in design or operation and such other matters as the PRC may assign to identify issues significant to nuclear safety and recommend nuclear safety improvements.

6.8.3.2 COMPOSITION

The Plant Safety and Licensing staff responsible for the review function shall be an experienced technical staff meeting the qualifications of Section 6.3.

6.8.3.3 RESPONSIBILITIES

The Plant Safety and Licensing staff may provide nuclear safety review as delegated by PRC for:

a. Procedures, programs and changes thereto identified in Specifications 6.4 and 6.5 and any additional procedures and changes thereto identified by the Plant General Manager as significant to nuclear safety.

b. All proposed tests or experiments.

c. All proposed changes or modifications to plant systems or equipment.

d. The Site Emergency Plan.

6.8.3.4 AUTHORITY

The Plant Safety and Licensing staff shall determine those issues significant to nuclear safety which require review by the Plant Review Committee from items considered under Specification 6.8.3.3.a through d. For those items not referred to PRC, Plant Safety and Licensing shall recommend in writing to plant management approval or disapproval of items considered under 6.8.3.3.

6.8.3.5 RECORDS

Reports of Plant Safety and Licensing activities pursuant to Specification 6.8.3.3 shall be submitted monthly to PRC.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated December 11, 1995, as supplemented by letters dated January 18, September 3, and October 2, October 18, and October 25, 1996, Consumers Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-20 for the Palisades Plant. The proposed amendment would modify TS Section 6.0 by removing or relocating requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS. The proposed changes would also revise other TS associated with the Section 6.0 revisions. Guidance on the proposed changes was developed by the NRC and provided in the Standard Technical Specifications (STS) for Combustion Engineering Plants, NUREG-1432, and Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," issued on December 12, 1995.

In addition to the proposed changes described above, the licensee also proposed to revise containment leak rate testing requirements to allow the Type A integrated leak rate test (ILRT) to be scheduled in accordance with Option B of 10 CFR Part 50, Appendix J.

The October 2, October 18, and October 25, 1996, letters provided clarifying information and updated TS pages that were within the scope of the initial application and did not affect the staff's initial proposed no significant hazards consideration determination, which was published following the September 3, 1996, submittal.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

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The Commission provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain LCOs, and related surveillance requirements and action statements, could be relocated from the TS to licensee-controlled documents. In *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979), the Atomic Safety and Licensing Appeal Board indicated that, "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

The Commission recently adopted amendments to 10 CFR 50.36, pursuant to which the rule was revised to codify and incorporate four criteria to be used in determining whether a particular matter is required to be included in an LCO, as follows: Final Rule, "Technical Specifications," 60 FR 36593 (July 19, 1995) (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; or (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. While the criteria specifically apply to LCOs, in adopting the revision to the rule the Commission noted that the staff had used the intent of these criteria to identify the optimum set of administrative controls in the TS (60 FR 36957).

10 CFR 50.36 states that "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The specific content of the administrative controls section of the TS is therefore that information that the Commission deems essential for the safe operation of the facility that is not already adequately covered by other regulations. Accordingly, the staff has determined that requirements that are not specifically required under 10 CFR 50.36(c)(5) and that are not otherwise necessary for operation of the facility in a safe manner, can be removed from administrative controls.

3.0 DESCRIPTION AND EVALUATION OF PROPOSED CHANGES

The licensee proposed to extensively restructure TS Section 6, "Administrative Controls," to closely emulate the Administrative Controls section of the STS. Certain requirements that are redundant to regulatory requirements would be deleted, other requirements would be relocated to licensee-controlled procedures and programs, and the remaining requirements would be reorganized to closely emulate the STS Administrative Controls section. The licensee also proposed changes to several requirements contained in other sections of the TS that were affected by this restructuring and proposed adoption of Option B of 10 CFR Part 50, Appendix J, for performance of Type A containment leak rate testing.

3.1 Requirements Deleted From Technical Specifications or Relocated To Other Licensee-Controlled Documents

Definitions

The licensee proposed to delete the definitions for MEMBER(S) OF THE PUBLIC, SITE BOUNDARY, and UNRESTRICTED AREA. Where these defined terms appear in the balance of the TS in capitalized text to indicate a term defined in Section 1, they would be replaced with lower case text. No requirements would be changed. The subject definitions are redundant to definitions contained in 10 CFR Part 20. There is no safety benefit to retaining these definitions in TS; therefore, the proposed change is acceptable.

Inservice Inspection Program Evaluation

The licensee proposed to delete TS 4.3e, which requires reevaluation of the Inservice Inspection Program in accordance with 10 CFR 50.55a(g)(5). This requirement is redundant to the referenced section of 10 CFR Part 50, and is included in the Inservice Inspection and Testing Program. The staff concludes that the regulatory requirements under 10 CFR 50.55a and the requirements of the Inservice Inspection and Testing Program provide sufficient control over reevaluation of the Inservice Inspection Program, such that removing these provisions from the TS is acceptable.

Plant Staff

The licensee has proposed that requirements contained in Table 6.2-1, "Minimum Shift Crew Composition," which are redundant to the requirements of 10 CFR 50.54(k), (l), and (m) be deleted. The regulations describe the minimum shift composition for operating modes, as well as for cold shutdown and refueling. The staff concludes that the shift staffing requirements in 10 CFR 50.54, in conjunction with the organizational responsibilities and authorities retained in the administrative controls, provide adequate control over the plant staffing requirements. Therefore, removing these requirements from the TS is acceptable.

Designation of Radiation Safety Manager

TS 6.3.2 would be revised to delete the requirements that the radiation safety manager be designated by the Plant General Manager, and that the radiation safety manager shall have direct access to the plant manager. In addition, the associated footnote, regarding clarification of the meaning of "Equivalent" as used in Regulatory Guide 1.8, would be incorporated into the main paragraph. The licensee stated that neither of the requirements regarding lines of responsibility between the Plant General Manager and the radiation safety officer are germane to a section on qualifications. The staff verified that these requirements are currently contained in the final safety analysis report (FSAR) section 12.1. The staff concludes that regulatory requirements of 10 CFR 50.59 and Part 50, Appendix B, provide adequate control of changes to these requirements. Therefore, removing these requirements from the TS is acceptable.

Safety Limit Violation

The licensee proposed to delete the requirements of existing TS 6.7, "Safety Limit Violation." As discussed in section 3.1 of this safety evaluation (SE), the Safety Limit violation shutdown requirement of TS 6.7.1a, would be moved to TS 2.1.1 and 2.2.1, which currently reference TS 6.7. TS 6.7.1.b, regarding 1-hour notification, and TS 6.7.1c and 6.7.1d, regarding written reporting would be deleted. The licensee stated that the TS 6.7.1.b requirement is redundant to 10 CFR 50.72(b)(1)(i)(A); and the requirements of TS 6.7.1.c and 6.7.1.d are redundant to 10 CFR 50.36(c)(1)(i)(A) and to 10 CFR 50.73(a)(2)(i)(A), (i)(B), and (ii)(B). This change would extend the required time for submitting a Safety Limit violation report from 14 days, as required by TS 6.7.1c and d, to 30 days, as required by 10 CFR 50.73. Since plant operation may not resume until authorized by the Commission, the staff finds that the extended reporting time would not have a significant impact on safety and is therefore acceptable.

Radiological Environmental Monitoring Program

The licensee proposed to delete the requirements of TS 6.8.4.b related to the Radiological Environmental Monitoring Program. The existing Radiological Environmental Monitoring Program requires that procedures be prepared for monitoring the radiation and radionuclides in the environs of plants, consistent with the guidance specified in 10 CFR Part 50, Appendix I. These procedures were developed to ensure that radioactive effluents are restricted to levels as low as reasonably achievable and have no impact on plant nuclear safety. The licensee has proposed to relocate the details and description of the program to the Offsite Dose Calculation Manual (ODCM), as was described in GL 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of Technical Specifications and Relocation of Procedural Details of RETS to the Offsite Dose Calculational Manual or the Process Control Program." The staff concludes that the requirements in 10 CFR 20.1302, 40 CFR Part 190, and 10 CFR Part 50, Appendix I, provide sufficient control of these provisions, and removing them from the TS is acceptable.

Startup Report

The licensee proposed to delete TS 6.9.1a, Startup Report. The existing TS requires that a summary of plant startup and power escalation testing be submitted within 90 days following completion of the start-up test program following amendment to the license involving a planned increase in power level, installation of fuel that has a different design or has been manufactured by a different fuel supplier, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. This reporting requirement simply summarizes plant records that are available for NRC review. The 90 days of operation allowed before submittal of the report and the lack of any required approval indicate that the report is not necessary for safe operation of the facility. This requirement is not specifically required under 10 CFR 50.36(c)(5) and the staff has determined that it is not otherwise necessary for operation of the facility in a safe manner. Therefore, this requirement can be removed from the administrative controls.

Reportable Events and Special Reports

The licensee proposed that the TS 6.9.2 requirement that the Commission be notified of all reportable events be deleted from the TS on the basis that this requirement is adequately addressed in the regulations. Requirements are provided in 10 CFR 50.73(a)(2) for the licensee to submit a Licensee Event Report (LER) for all reportable events specified in 10 CFR 50.73. The licensee also proposed to delete TS 6.9.4.b, which requires that special reports shall be submitted in accordance with 10 CFR 50.4, within the time period specified for each report. This TS merely repeats the requirement contained in 10 CFR 50.4. The staff concludes that the regulations contained in 10 CFR Part 50 provide sufficient control over these reporting requirements. Therefore, removing these requirements from the TS is acceptable.

Nonroutine Reports

The licensee proposed to delete TS 6.9.3, "Nonroutine Reports." Part (a) of the requirement requires submittal of a report in the event that the Radiological Environmental Monitoring Programs are not substantially conducted as described in the ODCM. Part (b) of the requirement dictates submittal of a report in the event of an occurrence that causes or could have caused a significant environmental impact.

The licensee stated that part (a) is not redundant to any other requirement, but plant administrative procedures require initiation of a Condition Report for such an event. Condition Reports are available for audit by the NRC resident inspector. This reporting requirement is not included in the STS, which reflects current staff position on those requirements which the staff considers to be necessary for safe operation of the facuility. This requirement is not specifically required under 10 CFR 50.36(c)(5) and the staff has determined that it is not otherwise necessary for operation of the

facility in a safe manner. Therefore, this requirement can be removed from the administrative controls.

The licensee stated that part (b) is redundant to Subsection 4.1 of Appendix B, "Environmental Protection Plan (Non-Radiological)," of the Palisades Facility Operating License. Appendix B Subsection 4.1 requires that any event that could or did result in significant environmental impact shall be promptly reported to the NRC within 24 hours, followed by a written report within 30 days. Because the information is duplicated in other required reports, the staff has concluded that the TS 6.9.3(b) requirement to submit environmental impact reports is unnecessary and deletion of this requirement is acceptable.

Record Retention

The requirements of Section 6.10, "Record Retention," would be deleted from the TS and relocated to the Quality Program Description, CPC-2A. There are no comparable Record Retention requirements in STS. The licensee proposed that the requirements be relocated because they are adequately addressed by CPC-2A, which implements 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The provisions in CPC-2A implement the Commission's regulations pertaining to the maintenance of records related to activities affecting quality contained in Criterion XVII, "Quality Assurance Records," of 10 CFR Part 50, Appendix B. Record retention requirements are specified in various regulations (e.g. 10 CFR Part 20, Subpart L, and 50.71). The staff has determined that record retention requirements are adequately addressed by the existing regulations and the related CPC-2A commitments relocated from the TS. The licensee submitted a revision to CPC-2A on May 31, 1996, which incorporated the record retention requirements into Sections 1, 2, and 3 of Appendix E to CPC-2A. The staff has verified that the record retention requirements have been incorporated without change into CPC-2A. Future changes to CPC-2A which would reduce the effectiveness of these commitments will be subject to prior review and approval by the NRC in accordance with 10 CFR 50.54(a).

The staff concludes that the regulatory requirements under 10 CFR Part 50, Appendix B, provide sufficient control of the plant records, and sufficient regulatory controls exist for future changes to the program pursuant to 10 CFR 50.54(a). In addition, other regulations (e.g. 10 CFR Part 20, Subpart L, and 10 CFR 50.71) require the retention of certain records related to operation of the plant. The staff concludes that these other regulatory requirements provide sufficient control of these recordkeeping provisions and removing them from the TS is acceptable. These requirements are not specifically required under 10 CFR 50.36(c)(5), and the staff has determined that they are adequately covered by other regulations. Therefore, these requirements can be removed from administrative controls.

Radiation Protection Program

The licensee proposed to remove the requirements in TS 6.11 related to the Radiation Protection Program. The existing TS for the Radiation Protection

Program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR Part 20. The requirement for procedures to implement Part 20 is also contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed under 10 CFR 20.1101(c).

The staff concludes that the requirements of the regulations provide sufficient requirements for the Radiation Protection Program, such that administrative controls in the TS are unnecessary. On this basis, the staff concludes that the requirements for the Radiation Protection Program do not have to be controlled by TS, and changes to the Radiation Protection Program are adequately controlled by 10 CFR Part 20, 50.54, 50.59, and Part 50, Appendix B. Therefore, the proposed change is acceptable.

Process Control Program

The requirement of existing TS 6.19 for Plant Review Committee (PRC) review and approval prior to Process Control Program changes becoming effective would be deleted. This requirement is redundant to the requirement of proposed TS 6.8.1.6a for PRC review and approval of changes to programs specified in TS 6.5. As discussed in section 3.4 of this SE, the remaining requirements of TS 6.19 would be relocated to TS 6.5.15. As a result of this relocation, the requirement for PRC review and approval of program changes contained in TS 6.19 would become redundant to the requirement of TS 6.8.1.6a. The staff concludes that the TS 6.8.1.6a requirement provides sufficient control of PRC review and approval of changes to the Process Control Program. Therefore, removing this requirement from the TS is acceptable.

Sealed Source Contamination

The existing requirement of TS 6.21 for a sealed source contamination program specifies limitations on fixed contamination for sources requiring leak testing, and states that sealed sources containing radioactive material shall be free of specified levels of removable contamination. The associated actions require that if the removable contamination exceeds limitations, the sealed source shall be either decontaminated or disposed of. The provisions will be relocated to the ODCM and the procedures regarding control of sealed source contamination will be adequately controlled in accordance with 10 CFR 50.59 and Part 20. These requirements are not specifically required under 10 CFR 50.36(c)(5) and are not necessary for operation of the facility in a safe manner. In addition, the staff has determined that the requirements are adequately covered by other regulations. Therefore, the requirements can be removed from administrative controls.

Summary

The existing TS requirements that would be deleted or relocated do not meet the intent of the four criteria described in 10 CFR 50.36(c)(2), and therefore are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. The requirements are not specifically required under 10 CFR 50.36(c)(5) and are adequately covered by other regulations. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59

and 50.54(a) to control future changes to relocated requirements. These requirements are not specifically required under 10 CFR 50.36(c)(5) and the staff has determined that they are not otherwise necessary for operation of the facility in a safe manner. Therefore, the staff has concluded that these requirements may be deleted or relocated from the TS to the specified documents.

3.2 Containment Leak Rate Testing Requirements

By supplemental application dated January 18, 1996, the licensee requested changes to permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Containment Leak Rate Testing Program," and proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

3.2.1 Background

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, Performance-Based Requirements, to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Palisades TS.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.2.2 Evaluation

The licensee's January 18, 1996, letter to the NRC proposes to establish a Containment Leakage Rate Testing Program and proposes to add this program to the TS. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to existing TS 4.5.1, "Containment Tests," and the addition of TS 6.5.14, "Containment Leak Rate Testing Program." Option B permits a licensee to choose Type A, or Type B and C, or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A testing on a performance basis. The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163 and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

3.3 Additional Proposed TS Requirement Changes

The licensee proposed several changes to TS requirements not directly related to the proposed administrative controls revision. These changes were proposed

as part of this amendment request because related specifications would be affected by the proposed administrative controls changes.

TS Table 4.2.3

The licensee proposed several revisions to the TS Table 4.2.3 testing requirements for the control room and fuel pool area ventilation systems. The existing requirement to verify that the control room ventilation system maintains the control room at a positive pressure of 0.1-inch water gauge (W.G.) would be revised to increase the required differential pressure capability to 1/8-inch (0.125-inch) W.G. and would require this differential pressure to be maintained with respect to the outside atmosphere rather than the viewing gallery. The existing requirement to verify control room temperature is less than or equal to 120 degrees F each 12 hours would be revised to require that the control room temperature be maintained less than or equal to 90 degrees F. Requirement 2.c would be added to Table 4.2.3, requiring verification each refueling cycle that the Fuel Pool Ventilation System is operable by initiation of flow through the HEPA filter from the control room.

The more restrictive requirements proposed by these changes are currently being administratively maintained by the licensee to provide assurance of ventilation system operability. The licensee stated that the revision of Table 4.2.3 made to support the revision of the Administrative Controls section provided a convenient opportunity to request revision of the existing requirements. These changes constitute more restrictive requirements which provide additional assurance that equipment conforms to the plant design basis and will operate reliably when called upon; therefore, the staff finds the proposed changes to be acceptable.

TS 4.6.2b

The licensee proposed to revise the TS 4.6.2b surveillance requirement and basis regarding verification that the containment spray nozzles are open to extend the surveillance frequency from 5 to 10 years. This requested change is consistent with the recommendation to extend this surveillance frequency contained in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," and GL 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." NUREG-1366 documented the staff's finding that the only problems of this nature have been construction-related and no such problems have been detected at any plants following completion of construction and startup testing. The licensee stated that there is no unusual feature of the Palisades spray nozzles that should require more frequent testing, and past testing at the 5-year interval has not identified any problems. The proposed change is consistent with staff guidance and is therefore acceptable.

TS 6.1.1

The licensee proposed to add a requirement to Section 6.1.1 for the plant superintendent to approve tests, experiments, and modifications. The

proposed wording is consistent with the STS. This change constitutes a more restrictive requirement that provides additional assurance that changes to plant procedures or equipment that could affect safety will receive an appropriate level of review prior to implementation, therefore the proposed change is acceptable.

TS 6.2.1

The licensee proposed to replace the specific titles of supervisory positions throughout the TS with generic titles and to revise TS 6.2.1a to require that the relationship between generic titles and plant-specific titles be included in the FSAR. The replacement of plant-specific titles with generic titles is an administrative change that does not result in a change in any TS requirements. The requirement to include the relationship between the TS generic titles and plant-specific titles in the FSAR will ensure changes to the titles will be adequately controlled in accordance with 10 CFR 50.59, and is therefore acceptable.

TS 6.7

The proposed change would add TS 6.7.3, which would allow high radiation areas with dose rates greater than 1000 mrem per hour inside large areas such as the containment, where no enclosure exists for purposes of locking, to be controlled by barricading and posting and identification with a flashing light. The current TS require that high radiation areas be locked. The licensee stated that construction of lockable enclosures solely for the purpose of bounding high radiation areas inside large areas such as the containment incurs both significant cost and personnel radiation exposure.

10 CFR Part 20.1601 requires strict controls for access to areas where an individual could receive greater than 0.001 Sv (100 mrem) in 1 hour (high radiation areas), providing reasonable assurance that individuals at licensed facilities are not exposed to radiation levels in excess of the regulatory limits. A list of optional controls for high radiation areas is provided in 10 CFR Part 20.1601(a) and (b). For large complex facilities, such as a nuclear power plant, the most practical of these options is to maintain the entrances to high radiation areas locked except during periods of authorized personnel access. In cases where the controls provided in 10 CFR 20.1601(a) and (b) unnecessarily restrict plant operations 10 CFR 20.1601(c) provides that the licensee may apply to the Commission for approval of alternative controls for the access of their high radiation areas.

Regulatory Position C.2.4 of Regulatory Guide 8.38, Revision 0, dated June 1993, describes acceptable alternative methods for controlling access to high radiation areas. Under this alternative control scheme, areas where individuals could receive doses in excess of 0.01 Sv (1000 mrem) in one hour, and which are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area, should be barricaded and conspicuously posted, with a flashing light activated as a warning device whenever the dose rate in such areas exceeds or is expected to exceed 0.01 Sv (1000 mrem) in 1 hour.

The proposed TS 6.7.3 requirement is consistent with the NRC's position on acceptable alternative methods for access control to high radiation areas in large areas as described in Regulatory Position C.2.4 of Regulatory Guide 8.38, Revision 0, and is consistent with the STS. Therefore, the proposed change is acceptable.

Summary

The proposed changes provide additional assurance that control room and fuel area ventilation equipment will operate reliably when called upon, revise the containment spray nozzle surveillance frequency to be consistent with current staff position regarding spray nozzle testing, provide additional assurance that changes to plant procedures or equipment will receive an appropriate level of review prior to implementation, ensure changes to the relationship between generic and plant-specific titles will be adequately controlled, and ensure that occupational doses continue to be maintained as low as reasonably achievable. These proposed changes are acceptable to the staff.

3.4 Editorial Restructuring

The licensee proposed editorial changes throughout the TS to support the proposed revision of the Administrative Controls section. The proposed changes primarily relocate TS requirements to, or within, the revised Administrative Controls section, or make minor editorial revisions. The proposed changes are described and evaluated individually in the following paragraphs.

TS 1.1 and 1.2

TS 1.1, "Operating Definitions," and 1.2, "Miscellaneous Definitions," would be combined into Section 1.0, "Definitions." The definitions for the Offsite Dose Calculation Manual and the Process Control Program would be combined with their respective program descriptions, which are moved to TS 6.5.1 and 6.5.15, respectively, as discussed under the TS 6.5 discussion, below. The definition for Core Operating Limits Report (COLR) would be arranged alphabetically with the remaining definitions. The reference in the COLR definition to the TS 6.9.1 COLR requirements would be revised to match the proposed relocation of these requirements to TS 6.6.5. These changes do not alter any TS requirements. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 2.1.1 and 2.2.1

The licensee proposed to move the requirement to shut down the reactor following a Safety Limit violation currently contained in TS 6.7.1a to TS 2.1.1 and 2.2.1. This change would place the operating requirement directly into the Action Statement, instead of incorporating it by reference. This is considered a more logical location from a human factors perspective. While this requirement is redundant to the noted Section of 10 CFR 50.36, it is retained in the TS to assure that operators are aware of the required action to immediately shut down the reactor if a Safety Limit is violated.

This is purely an administrative change in the location of the requirement within TS, and is therefore acceptable.

TS 3.17.4

The licensee proposed to revise TS 3.17.4, "Accident Monitoring Instrumentation," to move reporting requirements to the Administrative Controls section. TS 3.17.4.7 would be revised to reference the reporting requirement of TS 6.6.7, "Accident Monitoring Instrument Report." The proposed reporting requirement of TS 6.6.7 is unchanged from the existing TS 3.17.4.7 requirement. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 4.0.5

The requirements of TS 4.0.5 would be moved to Administrative Controls TS 6.5.7, "Inservice Inspection and Testing Program." The wording of the proposed specification would be editorially modified to be consistent with the STS; but there would be no change in requirements. References to TS 4.0.5 in the balance of the TS would be revised to reference TS 6.5.7. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS Table 4.2.3

The ventilation filter testing requirements of Table 4.2.3 would be replaced with a requirement to "Perform required Control Room Ventilation and Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program." The details of the TS Table 4.2.3 ventilation filter testing requirements would be moved to TS 6.5.10, "Ventilation Filter Testing Program." Existing surveillance requirement c.2 (proposed requirement 2.a) would be editorially revised from "a recirculation mode of operation" to "the emergency mode of operation" to more closely agree with current plant terminology. No requirements would be changed. These are purely administrative changes in the content or location of the requirements within TS, and are therefore acceptable.

TS 4.3f

TS 4.3f primary coolant pump testing requirements would be moved to proposed TS 6.5.6, "Primary Coolant Pump Flywheel Surveillance Program." There would be no change in primary coolant pump flywheel testing requirements. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 4.14

TS 4.14 would be retitled "Steam Generator Surveillance." The details of the testing requirements would be relocated to Administrative Controls as discussed below.

Existing TS 4.14.1, which currently requires the steam generators to be demonstrated operable by performance of the testing specified by TS 4.14.2 through 4.14.5 and TS 4.0.5 would be revised to require steam generator tube integrity to be verified in accordance with proposed TS 6.5.7, "Inservice Inspection and Testing Program," and proposed TS 6.5.8, "Steam Generator Tube Surveillance Program." The requirements of TS 4.14.2 through 4.14.5 would be moved to proposed TS 6.5.8, "Steam Generator Tube Surveillance Program." The requirements of TS 4.14.6 would be moved to proposed TS 6.6.9, "Steam Generator Tube Surveillance Report."

Table 4.14-1 would be eliminated, because the preservice and first inservice inspections have been completed and the requirements are no longer applicable. The Table 4.14-1 footnote, regarding performance of steam generator inspections on a rotating schedule, would be incorporated into the first paragraph of TS 6.5.8. The wording of the existing testing requirements would be revised to eliminate redundancies and to remove requirements pertinent only to preservice and initial testing.

The proposed testing program and reporting requirements are equivalent to the requirements of existing TS 4.14, with the exception of deletion of the preservice and initial inservice testing requirements, which have been completed and are therefore no longer required to be retained in TS. This is purely an administrative change in the content or location of the requirements within TS, and is therefore acceptable.

TS 4.16.1

The licensee proposed to revise the record retention requirement of TS 4.16.1f to eliminate reference to TS 6.10.2.1, which would be deleted by this amendment as discussed in section 3.1 of this SE. The TS still requires that records regarding snubbers be retained. As discussed in section 3.1 of this SE, existing regulatory requirements provide sufficient control of plant record retention. This is purely an administrative change, and is therefore acceptable.

TS 6.2.2

The requirements of TS Table 6.2-1, "Minimum Shift Crew Composition," which would not be deleted, as discussed in section 3.1 of this SE, would be included as separate requirements under TS 6.2.2. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 6.2.3

The licensee proposed to remove the reference to formerly deleted TS 6.2.3. The licensee stated that its retention served no function. This is purely an administrative change, and is therefore acceptable.

TS 6.3.4

TS 6.3.4 would be revised to delete reference to reviews performed in accordance with existing TS 6.5.3, and replace it with reference to reviews required by 10 CFR 50.59. The TS would be revised to allow the assignment of these reviews to any qualified plant staff. The TS currently requires reviews to be conducted by Plant Safety and Licensing staff. The qualification requirement for the individuals who perform the reviews would remain unchanged. The reference to existing TS 6.5.3 was reworded because this change request proposes deleting that section from TS as discussed in section 3.5 of this SE. The qualification requirements of the persons performing the subject reviews have not been changed, only the reference to their assignment in the plant organization. Therefore, the proposed changes are acceptable.

TS 6.3.5

TS 6.3.5 would be deleted. The requirements are redundant to other TS requirements either already present in existing TS or proposed by this amendment request. The first requirement, for an operations manager to hold an SRO license, and the third requirement, for the individual holding an SRO license to be responsible for directing activities of licensed operators, would be moved to proposed TS 6.2.2f. The second requirement, regarding Operations Manager qualifications of ANSI N18.1-1971, is redundant to existing TS 6.3.1 and would be deleted. No requirements would be changed. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 6.4

TS 6.4, which was deleted by a prior amendment, would be reestablished under a new title, "Procedures." Except as discussed for the security and emergency plans in section 3.1 of this SE, the existing procedure requirements of TS 6.8.1 would be moved to this section. TS 6.8.1a would be reworded slightly to emulate STS wording. With the exception of the deletion of the security and emergency plan requirements, no requirements would be changed. This is purely an administrative change in the location of the requirements within TS, and is therefore acceptable.

TS 6.5, Programs and Manuals

The review and audit requirements of TS 6.5 would be renumbered to TS 6.8 as discussed in section 3.5 of this SE, and TS 6.5 would be retitled "Programs and Manuals." Existing program and manual requirements contained in the Administrative Controls section, and new requirements proposed by this amendment, would be gathered in this section.

Existing Program Requirements Moved to TS 6.5

The licensee proposed to consolidate the following existing TS program requirements in proposed TS 6.5:

TS 6.8.4a, "Radioactive Effluent Controls Program," would be moved to TS 6.5.4. Editorial revisions would be made to make the wording consistent with the STS.

TS 6.15, "Systems Integrity," would be moved to TS 6.5.2, "Primary Coolant Sources Outside Containment." The requirements of TS 4.5.3, "Recirculation Heat Removal Systems," which contain additional testing requirements for systems outside the containment that could potentially contain highly radioactive fluids, would be incorporated into TS 6.5.2. The licensee stated that this change would consolidate all requirements for such testing in one location.

The requirements of TS 6.16, "Iodine Monitoring," and TS 6.17, "Post Accident Sampling," would be combined in proposed TS 6.5.3, "Post Accident Sampling Program."

TS 6.18, "Offsite Dose Calculation Manual," would be combined with its definition (moved from existing TS 1.2 as discussed in this SE) and would be moved to TS 6.5.1.

TS 6.19, "Process Control Program," would be combined with its definition (moved from existing TS 1.2 as discussed in this SE) and would be moved to TS 6.5.15. The existing requirement for PRC review and approval prior to Process Control Program changes becoming effective would be relocated to the Quality Program Description, CPC-2A, as discussed in section 3.1 of this SE.

The requirements of TS 6.22, "Secondary Water Chemistry," would be moved to TS 6.5.9, "Secondary Water Chemistry Program." The wording would be revised editorially to more closely match the STS.

None of the requirements of the existing TS consolidated under TS 6.5 would be changed, with the exception of editorial revisions for consistency with the STS and the change to the Process Control Program discussed in section 3.1 of this SE. These are purely administrative changes in the location of the requirements within TS, and are therefore acceptable.

<u>New Program Requirements</u>

The following program requirements would be added to TS 6.5, as previously discussed in this SE:

TS 6.5.6, "Primary Coolant Pump Flywheel Surveillance Program"
TS 6.5.7, "Inservice Inspection and Testing Program"
TS 6.5.8, "Steam Generator Tube Surveillance Program"
TS 6.5.10, "Ventilation Filter Testing Program"
TS 6.5.14, "Containment Leak Rate Testing Program"

TS 6.5.12, "Technical Specification Bases Control Program" would be added. The Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the TS Bases.

TS 6.5.5, would be reserved for future addition of the "Containment Structural Integrity Surveillance Program," as discussed in section 3.5 of this SE. TS 6.5.11 would be reserved for future addition of a "Fuel Oil Testing Program." TS 6.5.13 would be reserved for future addition of a "Safety Function Determination Program." No requirements would be added. This is purely an editorial change intended to reserve space for future additions to the Administrative Controls section to be consistent with the STS, and is therefore acceptable.

TS 6.6, Reporting Requirements

TS 6.9, "Reporting Requirements," would be moved to TS 6.6. Existing reporting requirements contained in the Administrative Controls section, and new requirements proposed by this amendment, would be gathered in this section.

Existing Reporting Requirements Moved to TS 6.6

The licensee proposed to consolidate the following existing TS reporting requirements in proposed TS 6.6:

> The requirements of TS 6.9.1b, "Annual Report," would be moved to TS 6.6.1, "Occupational Radiation Exposure Report."
>
> The requirements of TS 6.9.1c, "Monthly Operating Report," would be moved to TS 6.6.4.
>
> The requirements of TS 6.9.1d, "Radioactive Effluent Release Report," would be moved to TS 6.6.3.
>
> The requirements of TS 6.9.1e, "Radiological Environmental Operating Report," would be moved to TS 6.6.2.
>
> The requirements of TS 6.9.1f, "Core Operating Limits Report," would be moved to TS 6.6.5.
>
> The requirements of TS 6.9.4a, would be moved to TS 6.6.8, "Containment Structural Integrity Report."

The licensee proposed to add an option to the requirements of TS 6.6.1 to base radiation exposure reports on electronic dosimeters. Use of electronic dosimeters is equivalent to the other methods currently contained in the TS, as indicated by its inclusion in the STS, which reflects current staff position. The staff considers this to be an acceptable administrative change in the methods available for collecting and reporting radiation exposure data.

The licensee proposed to replace the reference in TS 6.6.1 to 10 CFR 20.407 with reference to 10 CFR 20.2206. The revised reporting requirement is equivalent to the existing reporting requirement of TS 6.9.1b. This is an administrative change to make the requirement consistent with the revised 10 CFR Part 20, and is acceptable.

None of the requirements of the existing TS consolidated under TS 6.6 would be changed, with the exception of editorial revisions for consistency with the STS and the revised 10 CFR Part 20. These are considered to be purely administrative changes in the content or location of the requirements within TS, and are therefore acceptable.

New Reporting Requirements

The following reporting requirements would be added to TS 6.6, as previously discussed in this SE:

TS 6.6.7, "Accident Monitoring Instrument Report"
TS 6.6.9, "Steam Generator Tube Surveillance Report"

TS 6.6.6 would be reserved for future addition of a "Pressure and Temperature Limits Report." This is purely an editorial change intended to reserve space for a future addition to the Administrative Controls to be consistent with the STS.

TS 6.7

Existing TS 6.7, "Safety Limit Violation,' would be deleted as discussed in section 3.1 of this SE, and TS 6.12, "High Radiation Area," would be moved to TS 6.7. TS 6.7.3 would be added as discussed in section 3.3 of this SE. TS 6.12.1 and 6.12.2 would be moved to TS 6.7.1 and 6.7.2, respectively. The wording of the proposed TS would be revised to be consistent with the STS. The TS 6.12.1 reference to the outdated 10 CFR 20.203(c) requirement would be revised to refer to the equivalent requirement of the revised 10 CFR Part 20, 10 CFR 20.1601. The updated requirements are equivalent to the existing 10 CFR Part 20 reference. No requirements of existing TS 6.12.1 and 6.12.2 would be changed. This is purely an administrative change in the location of the requirements within TS.

General Editorial Revisions

The licensee proposed the following general editorial changes to reformat the revised TS to eliminate voids and blank pages created as a result of the proposed changes. TS 4.5.6, "Containment Isolation Valves," would be renumbered to TS 4.5.3. The existing requirements of TS 4.5.3, would be relocated to TS 6.5.2 as discussed previously in this SE. The licensee proposed to revise wording throughout the Administrative Controls section to be consistent with the STS. The text of TS 4.2 through TS 4.6 and the entire Administrative Controls section would be repaginated to eliminate blank pages. Repagination would require that the list of modifying amendments, which appears at the bottom of each page, be rewritten to list only those amendments that modified the subject matter appearing on each proposed page. Existing pages 4-25 and 4-26 contain a notation at the bottom of the page that they were altered by "Change No. 16" and by "Amendment No. 12". The licensee stated that these are two identifiers for the same TS revision; "Change No. 16" to the TS was issued by "Amendment No. 12" to the Facility Operating License on February 25, 1975. The references to "Change No. 16" would be

omitted. The licensee also proposed to revise the TS Bases as necessary to be consistent with the proposed changes to the TS requirements.

Summary

The proposed changes relocate requirements within the TS, add requirements, and make minor editorial revisions to be consistent with the revised 10 CFR Part 20 and the STS. The preservice and initial inservice testing requirements have been completed and are therefore no longer required to be retained in TS. The use of electronic dosimeters is an acceptable alternative to measure personnel radiation exposure. Except as noted regarding the preservice and initial inservice steam generator inspections and the addition of the use of electronic dosimeters, no requirements would be changed. These are purely administrative and editorial changes, and are therefore acceptable.

3.5 Proposed Changes Still Under Staff Review

Review of the following changes proposed by the licensee has not yet been completed by the staff and will be the subject of subsequent licensing action.

The licensee proposed to relocate the requirements of TS 6.5 related to review and audit functions, TS 6.8.2 related to procedure review and approval, and TS 6.8.3 related to temporary procedure changes, to the Quality Program Description, CPC-2A. The licensee submitted a revision to CPC-2A on May 31, 1996, which was intended to incorporate the existing TS requirements. The staff identified several discrepancies between the existing TS requirements and the relocated requirements contained in the May 31, 1996, CPC-2A revision. The licensee will need to revise CPC-2A to correctly incorporate the TS requirements and submit the changes to the NRC for review prior to staff approval to delete these requirements from the TS.

In addition to the proposed relocation of the TS 4.3f primary coolant pump flywheel inspection requirement to the Administrative Controls section as discussed in section 3.4 of this SE, the licensee proposed to delete the TS 4.3f requirement to inspect the regenerative heat exchanger. This change, in concert with the relocation of the flywheel requirements, would permit deletion of TS Table 4.3.2, which is referenced by TS 4.3f and which contains the details of the inspections required by TS 4.3f. The licensee stated that this requirement is redundant to the ASME Boiler and Pressure Vessel Code, which requires the inspection to be performed once every 10 years. The existing requirement mandates that the inspection be performed at a maximum 5-year frequency. The staff requires additional information regarding the design codes and inspection results for the heat exchanger to complete review of this proposed change.

The licensee proposed to delete the requirements of TS 4.5.4, "Surveillance for Prestressing System," 4.5.5, "End Anchorage Concrete Surveillance," and 4.5.8, "Dome Delamination Surveillance," and replace the requirements with proposed TS 4.5.4, which would require verification of containment structural integrity in accordance with proposed TS 6.5.5, "Containment Structural

Integrity Surveillance Program," which would require containment tendons to be inspected in accordance with Regulatory Guide (RG) 1.35, Rev 3, July 1990. During review of the proposed change the staff identified that the RG does not address the dome delamination surveillances currently required by TS 4.5.8. Additional information will be necessary for the staff to complete its review of this proposed change.

The licensee proposed to revise limits on liquid and gaseous effluent release concentrations of proposed TS 6.5.4b and e, respectively, to adopt the requirements of 10 CFR Part 20, Appendix B, Table 2, in place of the former Appendix B, Table II limits. The licensee also proposed to revise related references to 10 CFR 20.106 contained in proposed TS 6.5.1c and 6.5.4c to refer to the revised requirement, 10 CFR 20.1301. During review, the staff identified that licensees proposing this revision typically adopt a requirement to require effluent concentrations to be maintained below *10 times* the 10 CFR Part 20, Appendix B, Table 2, limits, which is equivalent to the existing requirement. The licensee stated that this was its intent and it will submit a supplement to revise this request. The existing references to 10 CFR Part 20, Appendix B, Table II, and related references to 10 CFR 20.106, will be retained pending completion of staff review.

The licensee provided revised TS pages by letters dated October 18 and October 25, 1996, which incorporated the existing requirements into the revised TS pages. In order to accommodate the proposed restructuring of the administrative controls section as discussed in section 3.4 of this SE, the review and audit requirements of existing TS 6.5 have been renumbered as TS 6.8, and the requirements of existing TS 6.8.2 and 6.8.3 have been renumbered as TS 6.4.2 and 6.4.3, respectively. Several typographical errors in the resubmitted pages were corrected to properly reincorporate those items still under staff review. This was done to maintain the requirements in the TS while proceeding with the requested editorial restructuring of the TS.

Summary

These changes are under continued staff review. The NRC will issue a license amendment for proposed changes that are found to be acceptable upon completion of staff review. The page changes provided by letters dated October 18 and October 25, 1996, incorporate the existing TS requirements into the proposed revised TS pages. This is purely an administrative change to minimize the impact of the staff's continuing review of the requested changes on the proposed editorial repagination of the TS.

3.6 Conclusion

The staff has found that the existing TS requirements that would be deleted or relocated are appropriately governed by other regulatory requirements, do not meet the intent of the criteria described in 10 CFR 50.36(c)(2), are not specifically required under 10 CFR 50.36(c)(5), and are not otherwise necessary for operation of the facility in a safe manner. Sufficient regulatory controls exist to control future changes to relocated requirements.

With the exception of the proposed changes to TS requirements that remain under review by the staff, the containment leak rate testing and other TS requirement changes proposed by the licensee ensure continued conformance with the plant design bases and are consistent with the STS. The proposed editorial restructuring would administratively revise the TS requirements without affecting their technical content. Based on these considerations, the staff finds the proposed changes to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The Michigan State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 49493). The amendment also changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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