DCR
616

October 7, 1986

Docket No. 50-255

Correction to Amdt 70 to DPR-20

Mr. Kenneth W. Berry
Director, Nuclear Licensing
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. Berry:

During a recent administrative quality check of the Palisades Technical Specifications, a number of problem areas were identified. For the most part, these problems consisted of poor quality reproduction of TS pages resulting in illegible pages at the time of issuance of amendments, loss of text because of crowding material on a page, and a failure to update the Table of Contents as needed.

However, in two instances, amendments were issued which affected the same page and changes were dropped from one amendment to the other, thus restoring old text which had been deleted or changed with a previously-issued amendment. Those specific instances are as follows:

1. Amendment No. 68 was issued on December 8, 1981 and made changes to TS 3.10.3 on pages 3-58 and 3-59. Amendment No. 70 was issued on April 14, 1982 and made changes to TS 3.10.1 on page 3-58. The changes made by Amendment No. 68 were not incorporated with the changes made by Amendment No. 70, thus restoring material which had been changed by Amendment No. 68. A corrected page 3-58 is enclosed.

2. Amendment No. 85 was issued on November 9, 1984 and made changes to TS 6.5.1.6 on page 6-6. Amendment No. 86 was issued on December 10, 1984 and made an additional change to TS 6.5.1.6 and a change to TS 6.5.1.7. The previous change made by Amendment No. 85 was omitted from Amendment No. 86, thus restoring material which had been changed by Amendment No. 85. A corrected page 6-6 is enclosed. Amendment No. 85 instructed removal of a large number of pages; however, one page which should have been removed was overlooked, resulting in its remaining in the NRC Authority File copy. Page 6-17 issued November 12, 1975 is to be removed from the TS.

Amendment No. 25 was issued March 11, 1977 and made changes to TS 4.4a on page 4-24. One change to a number in the Basis was overlooked. Wherever the reference to "1835 psig" appeared, it should have been changed to "2135 psig". In one sentence on page 4-24, the change was not made. A corrected page is enclosed.

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On April 20, 1981, the Commission issued an Order with TS pages. On May 27, 1981, a letter with corrected TS pages 3-30 and 4-17 was sent to the licensee. The corrected pages were not incorporated into the Authority File copy at the time of issuance. Enclosed are pages 3-30 and 4-17 incorporating the footnote which was added on May 27, 1981.

As previously stated, a large number of TS pages are illegible or nearly so in the NRC Authority File, and the Table of Contents had not been updated on a number of occasions when amendments were issued which affected page numbers or section changes. Those TS pages which were of very poor quality have been retyped and the Table of Contents has been retyped in its entirety. In redoing the Table of Contents, the format was changed for easier reading, and three sections were expanded to list subsections which were omitted from the original listing.

Thank you for your cooperation in providing a copy of your master TS for comparison purposes. You will find that a number of pages affected were previously redone by your staff, thereby providing readable copy in your TS. Also, the Table of Contents enclosed is based on the pages as included in the NRC's Authority File and in some instances will not match the existing pages in your own TS. In cleaning up poor quality TS pages, your staff retyped and repositioned sections of the TS by adding "a" and "b" pages where required, resulting in different page numbers than contained in the NRC Authority File.

No changes have been made to the Technical Specifications as a result of this action. The amendment number shown on each page and the marginal lines indicating areas of change are those of the last action affecting the respective page.

An instruction sheet is included with the TS pages in order to further clarify this action. If you have any questions, please do not hesitate to contact me.

Sincerely,

/S/

Thomas V. Wambach, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
As stated

cc w/enclosures:
See next page

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Mr. Kenneth W. Berry
Consumers Power Company

Palisades Plant

cc:
M. I. Miller, Esquire
Isham, Lincoln & Beale
51st Floor
Three First National Plaza
Chicago, Illinois 60602

Mr. Thomas A. McNish, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Jerry Sarno
Township Supervisor
Covert Township
36197 M-140 Highway
Covert, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Palisades Plant
ATTN: Mr. Joseph F. Firlit
Plant General Manager
27780 Blue Star Memorial Hwy.
Covert, Michigan 49043

Resident Inspector
c/o U.S. NRC
Palisades Plant
27782 Blue Star Memorial Hwy.
Covert, Michigan 49043

Nuclear Facilities and
Environmental Monitoring
Section Office
Division of Radiological
Health
P.O. Box 30035
Lansing, Michigan 48909

INSTRUCTIONS

Remove pages	Insert pages
i-v Table of Contents	i-vi Table of Contents. Updated and reformatted for easier use. Surveillance Requirements 4.1, 4.5, 4.6 and 4.17 were expanded to include listing of subsections not previously listed.
2-3	2-3 NRC copy illegible. No change to text.
3-3	3-3 NRC copy lost text at bottom of page in reproduction. No change.
3-18	3-18 Repositioned to close gaps in text caused by deletions in middle of paragraph with last amendment issued. No change.
3-30 30-a	3-30 - Corrected as discussed in letter and repositioned to delete unnecessary "a" page which was nearly illegible.
3-58	3-58 Corrected as discussed in letter.
3-82	3-82 NRC copy lost text at right margin in production. No change to text.
3-88	3-88 NRC copy nearly illegible. No change to text.
3-102	3-102 NRC copy nearly illegible. No change.
4-17	4-17 Corrected as discussed in letter.
4-24	4-24 Basis corrected as indicated in letter. NRC copy nearly illegible.
4-30	4-30 NRC copy nearly illegible. No change.
4-36	4-36 NRC copy nearly illegible. No change.

Remove pages	Insert pages
4-40	4-40 NRC copy lost text at bottom of page in reproduction. No change.
4-70	4-70 NRC copy nearly illegible. No change.
4-71	4-71 NRC copy illegible.
4-72 4-72a	4-72 - NRC copy nearly illegible. Also deleted "a" page which was unnecessary.
4-74	4-74 NRC copy illegible. No change.
5-3	5-3 NRC copy nearly illegible. Repositioned to close gaps in paragraphs resulting from deletions of text with last amendment issued. No other changes.
6-6	6-6 Corrected as discussed in letter.
6-27	6-27 NRC copy nearly illegible. Text lost at bottom of page in reproduction.
6-28	6-28 NRC copy illegible.

PALISADES PLANT

TECHNICAL SPECIFICATIONS

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2.2 SAFETY LIMITS - PRIMARY COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on primary coolant system pressure.

Objective

To maintain the integrity of the primary coolant system and to prevent the release of significant amounts of fission product activity to the primary coolant.

Specification

The primary coolant system pressure shall not exceed 2750 psia when there are fuel assemblies in the reactor vessel.

Basis

The primary coolant system [1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The maximum transient pressure allowable in the primary coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the primary coolant system piping, valves and fittings under ASA Section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings and capacity of the secondary coolant system safety valves (985-1025 psig),[3] the reactor high-pressure trip ($\leq$2400 psia) and the primary safety valves (2500-2580 psia)[4] have been established to assure never reaching the primary coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psia (125% of design pressure) to verify the integrity of the primary coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves at 2400 psia and the secondary coolant system steam-dump and bypass valves at 900 psia.

References

(1) FSAR, Section 4.
(2) FSAR, Section 4.3.
(3) FSAR, SEction 4.3.4.
(4) FSAR, Section 4.3.9.

allowed during normal operation, so that substantial safety margin exists between this pressure differential and the pressure differential required for tube rupture.

Secondary side hydrostatic and leak testing requirements are consistent with ASME PBV Section XI (1971). The differential maintains stresses in the steam generator tube walls within code allowable stresses.

The minimum temperature of 100°F for pressurizing the steam generator secondary side is set by the NDTT of the manway cover of + 40°F.

The transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 126.9 x 10^6 lb/h minus 6% to account for flow measurement uncertainty and core flow bypass.[3] A steady state DNB analysis was also performed (assuming 115% overpower, 50 psi for pressure uncertainty, 3% for flow measurement uncertainty, and 3% for core flow bypass) in a parametric fashion to determine the core inlet temperature as a function of pressure and flow for which the minimum DNBR at 115% overpower is equal to 1.30.[4] The result of this steady state DNB analysis was the following equation for limiting reactor inlet temperature:

$$T_{inlet} \leq 541.0 + 0.03938\,(P-2060) + 0.00004843\,(P-2060)^2 + 1.0342\,(W-120.2)$$

A temperature measurement uncertainty of 3°F was subtracted from this limit in arriving at the LCO given in Section 3.1.1.g. The nominal full power inlet temperature is 2°F less than the value given in Section 3.1.1.g to allow for drift within the temperature control band. Thus, a total uncertainty of 5°F is applied to the limiting reactor inlet temperature equation. The limits of validity of this equation are:

$1850 \leq$ Pressure ≤ 2250 Psia

$110.0 \times 10^6 \leq$ Vessel Flow $\leq 130 \times 10^6$ Lb/h

The restrictions on starting a Reactor Coolant Pump with one or more PCS cold legs ≤ 250°F are provided to prevent PCS pressure transients, caused by energy additions from the secondary system, which would exceed the limits of Appendix G to 10 CFR Part 50. The PCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 70°F above each of the PCS cold leg temperatures.[5]

References
(1) FSAR, Sections 6.1.2.2 and 14.3.2.
(2) FSAR, Section 4.3.7.
(3) XN-NF-77-18.
(4) XN-NF-77-22.
(5) "Palisades Plant Overpressurization Analysis," June, 1977, and "Palisades Plant Primary Coolant System Overpressurization Subsystem Description," October, 1977.

3.1 PRIMARY COOLANT SYSTEM (Cont'd)

Basis

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the side boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power.

Permitting power operation to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram dose equivalent I-131, but below 40 μCi/gram, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation with specific activity levels exceeding 1.0 μCi/gram dose equivalent I-131 but below 40 μCi/gram must be restricted to no more than 10 percent of the unit's yearly operating time since the higher activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 15 following a postulated steam generator tube rupture.

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

The basis for the 40 μCi/gram limit is to provide margin for increased tube leakage as a result of a postulated steam line failure and to ensure that the consequences of this event would be within the 10 CFR Part 100 exposure guidelines.

(next page is 3-20)

3.3 EMERGENCY CORE COOLING SYSTEM (Cont'd)

g. A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the PCS cold legs is $\leq$ 250°F.

3.3.3 *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 and testing of Specification 4.3.h has not been accomplished in the previous 9 months, or prior to returning the check valves in Table 4.3.1 to service after maintenance, repair or replacement, the following conditions shall be met:

a. All pressure isolation valves listed in Table 4.3.1 shall be functional as a pressure isolation device, except as specified in b. Valve leakage shall not exceed the amounts indicated.

b. In the event that integrity of any pressure isolation valve specified in Table 4.3.1 cannot be demonstrated, at least two valves in each high pressure line having a non-functional valve must be in and remain in, the mode corresponding to the isolated condition.[1]

c. If Specifications a. and b. cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown condition within 12 hours, and cold shutdown within the next 24 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the primary coolant to near operating temperature by running the primary coolant pumps. The reactor is then made critical by withdrawing control rods and diluting boron in the primary coolant.[1] With this mode of start-up, the energy stored in the primary coolant during the approach to criticality is substantially equal to that during power operation and, therefore, all engineered safety features and auxiliary cooling systems are required to be fully operable. During low-temperature physics tests, there is a negligible amount of stored energy in the primary coolant; therefore, an accident comparable in

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*Effective after plant is placed in Refueling Shutdown Condition for 1981 refueling.

[1]Motor operated valves shall be placed in the closed position and power supplies deenergized.

Amendment No. 57, NRC Order dated April 20, 1981, corrected May 27, 1981

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to operation of control rods and hot channel factors during operation.

Objective

To specify limits of control rod movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.

Specifications

3.10.1 Shutdown Margin Requirements

a. With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%.

b. With less than four primary coolant pumps in operation at hot shutdown and above, boration shall be immediately initiated to increase and maintain the shutdown margin at $\geq$ 3.75%.

c. At less than the hot shutdown condition, boron concentration shall be shutdown boron concentration.

d. If a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable rod.

e. The drop time of each control rod shall be no greater than 2.5 seconds from the beginning of rod motion to 90% insertion.

3.10.2 Individual Rod Worth

a. The maximum worth of any one rod in the core at rated power shall be equal to or less than 0.6% in reactivity.

b. The maximum worth of any one rod in the core at zero power shall be equal to or less than 1.2% in reactivity.

3.10.3 Part-Lenth Control Rods

The part-length control rods will be completely withdrawn from the core (except for control rod exercises and physics tests).

3.18 Secondary Water Monitoring Requirements

Applicability:

Applies to the secondary water requirements of the steam generator blowdown during power operation (generator synchronized).

Objective:

To minimize potential steam generator tube degradation caused by contamination of the secondary coolant.

Specification:

3.18.1 Steam generator water requirements shall be maintained in accordance with Table 3.18.1 except as specified below.

3.18.2 The limits for pH and sodium specified in Table 3.18.1 shall be achieved within 72 hours of synchronization of the unit. If these limits are not established within this time, the reactor shall be brought to hot standby condition within an additional 12 hours.

3.18.3 During operation, other than that specified in 3.18.2, the limits for pH and sodium may exceed those specified in Table 3.18.1 for a period of up to 36 hours. If these limits are not restored, the reactor shall be placed in hot standby within an additional 12 hours.

3.18.4 The transient limit for specific conductivity specified in Table 3.18.1 shall be achieved within 72 hours of synchronization of the unit. If this limit is not established within this time period, then the reactor shall be brought to the hot standby condition within an additional 12 hours.

3.18.5 During operation, the steady-state limit for specific conductivity specified in Table 3.18.1 may be exceeded for a period of up to 7 days. If this limit is not restored within this time, the reactor shall be placed in hot standby within the next 12 hours.

3.20 Shock Suppressors (Snubbers)

Applicability

Applies to the operating status of the safety-related piping shock suppressors (snubbers) as listed in Tables 3.20.1 and 3.20.2.

Objective

To minimize the possibility of unrestrained pipe motion as might occur during an earthquake or severe transient.

Specification

3.20.1 During all modes of operation, except cold shutdown and refueling, all snubbers listed in Tables 3.20.1 and 3.20.2 shall be operable except as noted below:

a. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.16.1.c on the supported component or declare the supported system inoperable.

FIRE PROTECTION SYSTEM

3.22.5 PENETRATION FIRE BARRIERS

LIMITING CONDITIONS OF OPERATIONS

3.22.5.1 All penetration fire barriers protecting safety-related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

With one or more of the above required penetration fire barriers not intact a continuous fire watch shall be established on at least one side of the affected penetration within 1 hour. If an operable fire detector is located in the area, an hourly inspection of the penetration fire barrier may be performed rather than establishing a continuous fire watch.

BASIS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

inspection techniques that have been proven practical, and the conclusions of the evaluation shall be used as appropriate to update the inspection program.

f. Surveillance of the regenerative heat exchanger and primary coolant pump flywheels shall be performed as indicated in Table 4.3.2.

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. The specimen removal schedule shall be as indicated in Table 4.3.3.

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 high pressure line shall be recorded daily.

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* Effective after plant is placed in Refueling Shutdown Condition for 1981 refueling.

(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 to Table 4.3.1). Minimum test differential pressure shall not be less than 150 psid.

Amendment No. 33, NRC Order dated April 20, 1981, corrected May 27, 1981

4.4 PRIMARY COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for primary coolant system integrity.

Objective

To specify test for primary coolant system integrity after the system is closed following normal opening, modification or repair.

Specifications

a. Whenever the primary coolant system is closed after it has been opened, the system shall be leak tested at not less than 2135 psig prior to the reactor being made critical. A test temperature shall be selected such that secondary (saturation) pressure will limit the differential pressure across the steam generator tubes to not greater than 1380 psi.

b. Whenever modifications or repairs are made in the primary coolant system that involve new strength welds on components greater than 2-inch diameter, the new welds shall receive both a surface and 100% volumetric examination and shall meet all applicable code requirements.

c. Whenever modifications or repairs are made in the primary coolant system that involve new strength welds on components 2-inch diameter or smaller, the new welds shall receive a surface examination.

Basis

For normal opening, the integrity of the primary coolant system, in terms of strength, is unchanged. If the system does not leak at 2135 psig (operating pressure + 50 psi; ± 50 psi is normal system pressure fluctuation)[1], it will be leak tight during normal operation. If the pressure goes above 2135 psig, the worst consequence is a leak.

Calculations have been performed to demonstrate that a pressure difference of 1380 psi can be withstood by a tube uniformly thinned to 36% of its original nominal wall thickness (64% degradation), while maintaining:

a) A factor of safety of three between the actual pressure differential and the pressure differential required to cause bursting.

b) Stresses within the yield stress for Inconel 600 at operating temperature.

c) Acceptable stresses during accident conditions.

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4.5.5 End Anchorage Concrete Surveillance

a. Specific locations for surveillance will be chosen from the combined information from the design calculations; the as-built end anchorage concrete and prestressing records; observations of the end anchorage concrete during and after prestressing; and the results of strain and deformation measurements made during prestressing and the initial structural test.

b. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

c. The inspections made shall include:

   (1) Visual inspection of the end anchorage concrete exterior surfaces.
   (2) A determination of the temperatures of the liner plate area or containment interior surface in locations nearest to the end anchorage concrete under surveillance.
   (3) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.
   (4) The mapping of the predominant visible concrete crack patterns.

water to the containment sump inventory can be readily accommodated via the charging pumps from either the SIRW tank or the concentrated boric acid storage tanks.

In case of failure to meet the acceptance criteria for leakage from the shutdown cooling system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor. The times allowed for repairs are consistent with the times developed for other engineered safety feature components.

A reduction in prestressing force and changes 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 18
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, an investigation shall be made to detect and correct the causes.[6,7]

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

4.6.4 Valves

Deleted

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.

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 annual 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 in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1, 2]

4.15 Primary System Flow Measurement

Applicability

Applies to the measurement of primary system flow rate with four primary coolant pumps in operation.

Objective

To provide assurance that the primary system flow rate is equal to or above the flow rate required in 3.1.1(c).

Specification

After each refueling outage, or after plugging 10 or more steam generator tubes, a primary system flow measurement shall be made with four primary coolant pumps in operation before the reactor is made critical.

Basis

This surveillance program assures that the reactor coolant flow is consistent with that assumed as the basis for Specification 3.1.1(c).

4.16 Inservice Inspection Program for Shock Suppressors (Snubbers)

Applicability

Applies to periodic surveillance of safety-related snubbers as listed in Tables 3.20.1 and 3.20.2.

Specification

4.16.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

a. Visual Inspection

Visual inspection shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.16.1.d or 4.16.1.e, as applicable. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

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

#The provisions of Specification 4.0.2 are not applicable.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (10% of the total safety-related snubbers in use at the plant) shall be functionally tested either in place or in a bench test. The test shall verify the snubber has freedom of movement and is not frozen up. For each snubber which did not meet the functional test acceptance criteria of Specification 4.16.1.d or 4.16.1.e, an additional 10% of the total shall be functionally tested.**

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. Snubbers identified in Table 3.20.1 and 3.20.2 as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.* Table 3.20.1 and 3.20.2 may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are suppressed by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components suppressed by the snubber(s) were adversely affected by the inoperability of the snubbers in order to ensure that the suppressed component remains capable of meeting the designed service.

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*Permanent or other exemptions from functional testing for individual snubbers may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

**Functional tests for snubbers of rated capacity greater than 50,000 pounds will not be performed until such time that suitable on-site testing equipment is available. In the interim, a stroke test will be performed to verify freedom of movement over the full range of stroke in both compression and tension.

Basis

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

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

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Cont'd)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of about 136 inches and an active height of about 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 assemblies. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution.

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4. Poison may be placed in the fuel bundles for long-term reactivity control.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed boron rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-5 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3. Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, two high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

RESPONSIBILITIES (Continued)

c. Review of all proposed changes to Appendix "A" Technical Specifications.

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

e. Investigation of all violations of the Technical Specifications. (A report shall be prepared covering evaluation and recommendations to prevent recurrence and forwarded to the Vice President - Nuclear Operations and to the Executive Engineer - NAPO.)

f. Review of plant operations to detect potential nuclear safety hazards.

g. Performance of special reviews and investigations and reports thereof as requested by the Plant General Manager or Chairman of NSB.

h. Review of the Site Emergency Plan and implementing procedures.

i. Review of all reportable events as defined in Section 1.4.

PRC review may be performed through a routing of the item subject to the requirements of Specification 6.5.1.7.

6.5.1.7 AUTHORITY

The PRC shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Specifications 6.5.1.6.a through d. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6.a through e. above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President - Nuclear Operations and to the Vice Chairman of NSB of any disagreement 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 Specifications 6.5.1.6.a. through d. above based on a routine 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 recommen-

e. Records of training and qualification for current members of the plant staff.

f. Records of reactor tests and experiments.

g. Records of changes made to Operating Procedures.

h. Records of radioactive shipments.

i. Records of sealed source leak tests and results.

j. Records of annual physical inventory of all source material of record.

k. Chlorine treatment records.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of monthly radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

f. Records of inservice inspections performed pursuant to these Technical Specifications.

g. Records of Quality Assurance activities required by the QA Manual to be retained for the duration of the facility operating license.

h. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

i. Records of meetings of the PRC and SARB.

j. Records of monthly facility radiation and contamination surveys.

k. Records for environmental qualification which are covered under the provisions of paragraph 6.14.

l. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.20.1 and 3.20.2 including the date at which the service life commences and associated installation and maintenance records.

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6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR, Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hour but less than 1000 mrem/hour 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*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.