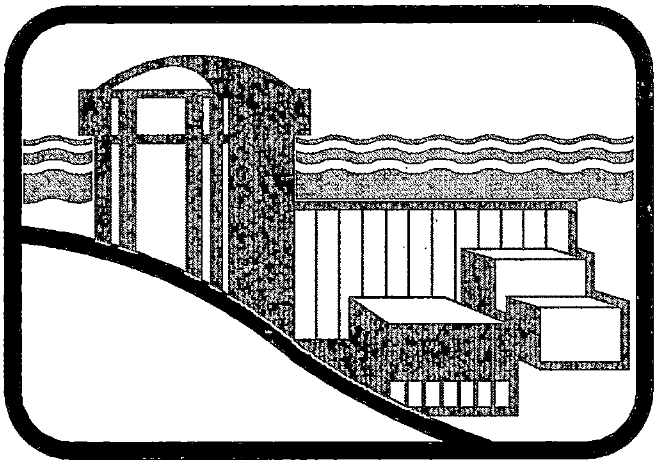
IMPROVED TECHNICAL SPECIFICATIONS



ALISADES
UCLEAR
LANT

Volume 20 **CHAPTER 5.0**

Consumers Energy

PALISADES NUCLEAR PLANT
CONSUMERS ENERGY
Docket 50-255
Conversion to Improved Technical Specifications
License DPR-20

INTRODUCTION: CHAPTER 5.0, ADMINISTRATIVE CONTROLS

A. ARRANGEMENT AND CONTENT OF THIS CHAPTER OF THE CHANGE REQUEST

This Chapter of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specifications addressing Chapter 5.0, "ADMINISTRATIVE CONTROLS." These changes are intended to result in requirements which are appropriate for the Palisades Nuclear Plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1, Chapter 5.0.

This discussion and its supporting information frequently refer to three sets of Technical Specifications, and to two groups of discussions associated with the proposed changes; the following abbreviations are used for clarity and brevity:

CTS - The Palisades Current Technical Specifications,
ITS - The Palisades Improved Technical Specifications,
ISTS - NUREG 1432, Revision 1.
DOC - Discussions of Change; these discussions explain and justify the differences between the requirements of CTS and ITS.
JFD - Justifications for Deviation; these discussions explain the differences between the requirements of the ITS and the ISTS.

Six attachments are provided to assist the reviewer:

1. Proposed ITS Chapter 5.0 pages

2. This Attachment is not applicable to Chapter 5.0

3. A set of all those CTS pages which contain requirements associated with those in ITS Chapter 5.0, marked up to show the changes from CTS to ITS, and arranged by specification in the order in which the requirements occur in ITS. This attachment also includes a DOC for each change.

   Each change from CTS to ITS is classified in the following categories:

   ADMINISTRATIVE - A change which is editorial in nature, which only involves movement of requirements within the TS without affecting their technical content, or clarifies CTS requirements.

   MORE RESTRICTIVE - A change which only adds new requirements, or which revised an existing requirement resulting in additional operational restrictions.

   RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the CTS to the Operating Requirements Manual (which has been included in the FSAR by reference).

A. **ARRANGEMENT AND CONTENT OF THIS CHAPTER OF THE CHANGE REQUEST** (continued)

LESS RESTRICTIVE - REMOVAL OF DETAIL (LA) - A change in which certain details from otherwise retained Specifications are removed from the ITS and placed in the Bases, FSAR, or other licensee controlled documents.

LESS RESTRICTIVE - A change which deletes any existing requirement, or which revises any existing requirement resulting in reduced operational restrictions.

4. No Significant Hazards Analyses for the changes from CTS to ITS.

   An individual No Significant Hazards Analysis is provided for each Less Restrictive change; generic No Significant Hazards Analyses are provided for each of the other categories of change.

5. ISTS Chapter 5.0 marked to show the differences between ISTS and ITS.

6. JFDs for the differences between ISTS and ITS.

B. **REFERENCE DOCUMENTS**

This Chapter of the TSCR is based on the following reference documents:

1. CTS as revised through Amendment 178.

2. The following TSCRs which are currently under review by the NRC:

   a. Administrative Controls, initially submitted on December 12, 1995.

   b. Electrical, initially submitted on December 27, 1995.

   c. Containment, submitted on March 26, 1997.

   d. PCP Flywheel, initially submitted on January 18, 1996.

3. ISTS, as revised by Industry Generic Changes (TSTF) approved as of October 15, 1997.

4. The following changes to ISTS which are currently under review by the NRC:

   a. TSTF 52.

C. THE UNIQUE PALISADES NUCLEAR PLANT FEATURES AFFECTING THIS CHAPTER

Palisades has several physical, analytical, and administrative features which differ from those newer CE plants upon which the ISTS were based. Palisades was the first CE plant designed and built. Its design and licensing preceded the issuance of the General Design Criteria so that, in some aspects, its physical systems are not like those of newer plants; its Technical Specifications preceded the issuance of Standard Technical Specifications (STS) so that LCOs, Actions, and Surveillance Requirements are not coordinated as they would be for a STS plant. Palisades has purchased all its core reloads from Siemens Power Corporation (or its predecessors), therefore, reload analyses and the associated core physics parameters, as well as certain Safety Analyses are not like those plants using all CE fuel and analyses as were modeled in the ISTS.

D. THE DIFFERENCES BETWEEN CTS "OPERATING CONDITIONS" AND ITS "MODES"

The CTS definitions of plant operating conditions have been replaced with the operation Mode definitions used in ISTS. In several instances the name for a CTS defined "operating condition" is the same as that for an ISTS "Mode," but the definition differs.

CTS contain the following definitions for operating conditions:

1. 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.

2. 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.

3. 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.

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

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

D. THE DIFFERENCES BETWEEN CTS "OPERATING CONDITIONS" AND ITS "MODES"

ITS contain the following definition table for Modes:

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

E. MODE CHANGES USING CTS "OPERATING CONDITIONS" VERSUS ITS "MODES"

1. CTS "Power Operation" is essentially equivalent to ITS "MODE 1." Each represents a condition with the reactor critical and the turbine generator in operation. The only effective difference is the power level which separates that Condition or Mode from the next lower one. During plant startup, the plant must meet all CTS "Power Operation" or ITS "MODE 1" LCOs before the turbine generator is placed on the line; similarly, during plant shutdown, the plant exits CTS "Power Operation" or ITS "MODE 1" when the turbine generator is no longer in service. Therefore, this change in definition will have no operational effect.

2. CTS "Hot Standby" is similar to ITS "MODE 2." Each represents a condition with the reactor critical, or nearly so, and the turbine generator shut down. During plant startup, the plant must meet all CTS "Hot Standby" or ITS "MODE 2" LCOs before a reactor startup is started; during plant shutdown, the plant exits CTS "Hot Standby" or ITS "MODE 2" when the reactor is shutdown. CTS action statements requiring that the plant be placed in "Hot Standby" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 2." Therefore, this change in definition will have no operational effect.

E. MODE CHANGES USING CTS "OPERATING CONDITIONS" VERSUS ITS "MODES" (continued)

3. CTS "Hot Shutdown" and ITS "MODE 3" are similar at their upper temperature boundary. During plant shutdown, the plant exits CTS "Hot Standby" or ITS "MODE 2" when the reactor is shutdown. CTS action statements requiring that the plant be placed in "Hot Shutdown" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 3." CTS "Hot Shutdown" and ITS "MODE 3" are quite different at their lower temperature boundary; CTS "Hot Shutdown" is exited when Tave drops below 525°F, ITS "MODE 3" is not exited until Tave drops below 300°F.

4. CTS does not provide a defined term for the condition when Tave is between 525°F and 210°F (the upper bound for CTS "Cold Shutdown").

5. CTS "Cold Shutdown" is essentially equivalent to ITS "MODE 5." Each represents a condition with Tave below boiling. There is no technical significance to the difference between the CTS 210°F and the ITS 200°F. CTS action statements requiring that the plant be placed in "Cold Shutdown" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 5." Therefore, this change in definition will have no operational effect.

6. CTS "Refueling Shutdown" is essentially equivalent to ITS "MODE 6." Each, when taken with other definitions and LCO requirements, represents a condition with the reactor at least 5% shutdown. Therefore, this change in definition will have no operational effect.

F. THE MAJOR CHANGES FROM CTS (as modified by pending TSCRs) TO ITS

1. The Safety Function Determination Program, 5.5.13, has been added to ITS Chapter 5.0, but does not appear in CTS Chapter 6.0. The Safety Function Determination Program is a feature of ISTS which supports LCO 3.0.6 (which has also been added to ITS) in addressing support system operability.

G. THE MAJOR DIFFERENCES BETWEEN ITS AND ISTS

1. The Explosive Gas and Storage Tank Radioactivity Monitoring Program, ISTS 5.5.12, was omitted from the ITS. Palisades CTS contain no equivalent requirements.

2. The Fuel Oil Testing Program, ITS 5.5.11, contains different testing and sampling requirements than its ISTS counterpart. These differences are necessary because the Palisades Fuel Oil Storage Tank serves several components besides the Diesel Generators, and the turn over rate for the fuel oil is quite high. This results in different limiting conditions than a tank where fuel is stored for a long time.

3. The Containment Leak Rate Testing Program, ITS 5.5.14, has been revised to allow use of 10 CFR 50, Appendix J, Option B, performance based testing.

ATTACHMENT 1

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

PROPOSED TECHNICAL SPECIFICATIONS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 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.

5.1.2 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 in MODE 1, 2, 3, or 4, 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 MODE 5 or 6 an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 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.

5.2 Organization

5.2.2 Plant Staff

a. A non-licensed operator shall be assigned when fuel is in the reactor and an additional non-licensed operator shall be assigned when the reactor is operating in MODES 1, 2, 3, or 4.

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 plant is in MODES 1, 2, 3, or 4, 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 5.2.2.a and 5.2.2.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.

5.2 Organization

5.2.2 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 plant. If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed.

5.0 ADMINISTRATIVE CONTROLS

5.3 Plant Staff Qualifications

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

5.3.2 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.

5.3.3 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.

5.3.4 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.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Site Fire Protection Program implementation.

d. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

5.5.1 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 5.6.2. and Specification 5.6.3.

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 20.1302, 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.

5.5 Programs and Manuals

5.5.2 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).

5.5.3 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 gases 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.

5.5 Programs and Manuals

5.5.4 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 functional capability 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 times the value in 10 CFR 20, Appendix B, Table 2, Column 2.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 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 plant 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 times the value listed in 10 CFR 20, Appendix B, Table 2, 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 plant to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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.

5.5.5 Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL.

If, as a result of a tendon inspection, 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, 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 in accordance with Specification 5.6.7, "Containment Structural Integrity Surveillance Report."

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

5.5.6 Primary Coolant Pump Flywheel Surveillance Program

a. Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

b. The provisions of SR 3.0.2 are applicable to the Flywheel Testing Program.

5.5 Programs and Manuals

5.5.7 Inservice Testing Program

This program provides controls for inservice 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 SR 3.0.2 are applicable to the above required intervals for performing inservice testing activities;

c. The provisions of SR 3.0.3 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.

5.5.8 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.

5.5 Programs and Manuals

5.5.8 Steam Generator Tube Surveillance Program

a. Steam Generator Tube Sample Selection and Inspection (continued)

The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.8-1. 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.

3. The tubes selected as the second and third samples (if required by Table 5.5.8-1) 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.

5.5 Programs and Manuals

5.5.8 Steam Generator Tube Surveillance Program (continued)

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.

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.8-1 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 5.5.8.b.1; the interval may then be extended to a maximum of once per 40 months.

5.5 Programs and Manuals

5.5.8 Steam Generator Tube Surveillance Program (continued)

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.8-1 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 LCO 3.4.13.

   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.

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

5.5 Programs and Manuals

5.5.8 Steam Generator Tube Surveillance Program (continued)

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 5.5.8.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 5.5.8-1.

The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program.

TABLE 5.5.8-1
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION[1]		2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Result	Action Required	Result	Action Required	Result	Action Required
C-1	None	N/A	N/A	N/A	N/A
C-2	Plug defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG.	C-2	Plug defective tubes and inspect additional 4S tubes in this SG.	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
		All other SGs are C-1	None	N/A	N/A
		Some SGs C-2 but no other SG is C-3	Perform action for C-2 result of second sample	N/A	N/A
		Other SG is C-3	Inspect all tubes each SG and plug defective tubes	N/A	N/A

NOTES: 1 The minimum sample size for the first sample inspection is S tubes per SG where S=(6/n)%, where n is the number of steam generators inspected during an inspection.

5.5 Programs and Manuals

5.5.9 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.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation 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 Fuel Handling Area Ventilation System 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%

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (continued)

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 Fuel Handling Area Ventilation System 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 the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

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

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (continued)

The provisions of SR 3.0.2 and SR 3.0.3 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.

5.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

    1. API gravity or an absolute specific gravity,

    2. Kinematic viscosity, and

    3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5 Programs and Manuals

5.5.12 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 5.5.12.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).

5.5.13 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.14 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.

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

Local leak rate tests, other than Personnel Airlock doors between the seals tests, shall be performed at ≥ 55 psig.

5.5 Programs and Manuals

5.5.14 Containment Leak Rate Testing Program (continued)

Local leak rate tests for checking airlock doors seals within 72 hours of each door opening shall be performed as follows:

a. A between the seals test shall be performed on the Personnel Airlock at ≥ 10 psig.

b. A full pressure test shall be performed on the Emergency Escape Airlock at ≥ 55 psig. A seal contact check shall be performed on the Emergency Escape Airlock following each full pressure test. Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

Leak rate acceptance criteria are:

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

b. The leakage for a Personnel Airlock door seal test shall not exceed 0.023 L_a.

c. An acceptable Emergency Escape Airlock door seal contact check consists of a verification of continuous contact between the seals and the sealing surfaces.

Containment OPERABILITY is equivalent to "Containment Integrity" for the purposes of the air lock testing requirements in 10 CFR 50, Appendix J.

The provisions of SR 3.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

5.5 Programs and Manuals

5.5.15 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.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

This report shall include a tabulation on an annual basis of the number of stations, utility and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent greater than 100 mrem and the associated deep dose equivalent (reported in person-rem) 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 assignments to various duty functions may be estimated based on pocket ionization chamber, 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 deep dose equivalent external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the plant 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.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering operation of the plant in the previous year shall be submitted prior to May 1 of each year 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 plant. 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.

5.6 Reporting Requirements

5.6.4 Monthly Operating Report

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

5.6.5 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:

3.1.7	Regulating Rod Group Position Limits
3.2.1	Linear Heat Rate Limits
3.2.2	Radial Peaking Factor Limits
3.2.4	ASI 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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs 3.1.6, 3.2.1, & 3.2.2)

   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.

   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 3.1.6, 3.2.1, & 3.2.2)

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 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

10. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.2.1, 3.2.2, & 3.2.4)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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.

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 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.

5.6.7 Containment Structural Integrity Surveillance Report

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

5.6.8 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

5.6 Reporting Requirements

5.6.8 Steam Generator Tube Surveillance Report (continued)

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.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 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 $\geq$ 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., radiation safety 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.

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, except as allowed by 5.7.3, 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 radiation safety 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.

5.7.3 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.

ATTACHMENT 2

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

PROPOSED BASES (N/A for CHAPTER 5.0)

ATTACHMENT 3

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

CTS MARKUP

AND

DISCUSSION OF CHANGES

4.4 Deleted

5.0 ADmin Controls

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, other than Personnel Airlock doors between the seals tests, shall be performed at ≥ 55 psig.

(2) Local leak rate tests for checking airlock door seals within 72 hours of each door opening shall be performed as follows:

5.5.14

(a) A between the seals test shall be performed on the Personnel Airlock at ≥ 10 psig.

(b) A full pressure test shall be performed on the Emergency Escape Airlock at ≥ 55 psig. A seal contact check shall be performed on the Emergency Escape Airlock following each full pressure test. Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

(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:~~

(see 3.6)

(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.~~

5.0 ADMIN
CONTROLS

4.5.2 Local Leak Detection Tests (continued)

b. Acceptance Criteria

5.5.14

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

(2) The leakage for a Personnel airlock door seal test shall not exceed 0.023 L_a.

(3) An acceptable Emergency Escape Airlock door seal contact check consists of a verification of continuous contact between the seals and the sealing surfaces.

(See 3.6)

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 within 48 hours, the plant shall be placed in at least HOT SHUTDOWN within the next 6 hours and in 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 place the plant in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(3) If the Personnel airlock door seal leakage is greater than 0.023 L_a, or if the Emergency Escape Lock door seal contact check fails to meet its acceptance criterion, repairs shall be initiated immediately to restore the door seal to the acceptance criteria of specification 4.5.2.b(2) or 4.5.2.b(3). In the event repairs cannot be completed within 7 days, the plant shall be placed in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(4) If air lock door seal leakage results in one 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 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 placed in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Entry and exit is permissible through a "locked" air lock door to perform repairs on the affected air lock components.

4.5 CONTAINMENT TESTS

4.5.4 Surveillance for Prestressing System

5.0
5.5.5

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: LA.1

(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 CONTAINMENT TESTS

4.5.4 Surveillance for Prestressing System (continued)

5.0

5.5.5

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.

LA.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 CONTAINMENT TESTS

4.5.4 Surveillance for Prestressing System (continued)

5.0

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.

5.5.5

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

LA.1

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

5.5.5 →

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~~.~~ A.1

A.10 in accordance with Specification 5.6.7, "Containment Structural Integrity Surveillance Report".

ITS

5.0 6.0 ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY

5.1.1 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.

MODE 1, 2, 3 or 4

5.1.2 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. A.2

MODE 5 or 6

5.2 6.2 ORGANIZATION

5.2.1 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.

5.0 6.0 ADMINISTRATIVE CONTROLS

5.2.2 6.2.2 Plant Staff

WHEN FUEL IS IN THE

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~~ IN MODES 1, 2, 3, OR 4. WHEN THE A.3 A.2

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~~ PLANT IS IN MODES 1, 2, 3, OR 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. A.2 A.4

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.

ITS

5.0 6.0 ADMINISTRATIVE CONTROLS

5.2.2.e 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. 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. 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~~ plant. If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed. A.4

5.3 6.3 PLANT STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

5.3.2 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.

5.3.3 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.

5.3.4 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.

5.0 6.0 ADMINISTRATIVE CONTROLS

5.4 6.4 PROCEDURES

5.4.1 6.4.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

A.1

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.~~ H.5

c ~~d.~~ Site Fire Protection Program implementation.

d ~~e.~~ All programs specified in Specification 6.5.

~~f. Site Security Plan implementation.~~

~~g. Site Emergency Plan implementation.~~ A.6

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. M.1

5.0 6.0 ADMINISTRATIVE CONTROLS

5.5 6.5 PROGRAMS AND MANUALS

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

5.5.1 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~~ 5.6.2. and Specification ~~6~~ 5.6.3. A.1

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 20.1302, 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.

5.0 ## 6.0 ADMINISTRATIVE CONTROLS

5.5.2 ### 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).

5.5.3 ### 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
gases coolant, radioactive iodines and particulates in plant gaseous M.2
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.

5.0 6.0 ADMINISTRATIVE CONTROLS

5.5.4 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 Functional capability ~~operability~~ of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM, A.1

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

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM, A.1

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 Plant ~~unit~~ to unrestricted areas conforming to 10 CFR 50, Appendix I, A.1

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 times the value listed in 10 CFR 20, Appendix B, Table 2, 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.

5.5.5 6.5.5 ~~Reserved~~ CONTAINMENT STRUCTURAL INTEGRITY SURVEILLANCE PROGRAM M.3

< ADD Program from NUREG-1432 >

5.5.6 6.5.6 Primary Coolant Pump Flywheel Surveillance Program

a. Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

b. The provisions of Surveillance Requirement 4.0.2 are applicable to the Flywheel Testing Program.

5.5.7 6.5.7 Inservice ~~Inspection and~~ Testing Program A.7

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~~ SR 3.0.2 are applicable to the above required intervals for performing inservice testing activities; A.1

c. The provisions of ~~Surveillance Requirement 4.0.3~~ SR 3.0.3 are applicable to inservice testing activities; and A.1

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

6.0 ADMINISTRATIVE CONTROLS

5.5.8 **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 manne . 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 5 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: A.1

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.

5.5.8 | 6.5.8 Steam Generator Tube Surveillance Program (continued)

(A.1)

3. The tubes selected as the second and third samples (if required by Table 6 5 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.

5.0 ADmin Controls

5.5.8 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. A.1

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.1

   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~~ LCO 3.4.13 A.1

   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.

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5.5.8 | 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 5.8.b.3, above. A.1

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 5.8-1. A.1

The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program. A.11

TABLE ~~6~~ 5.5.8-1
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. NOTE 1	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.	All other S.G.s are C-1	None	N/A	N/A
ITS 5.6.8		24 hour verbal notification to NRC with written follow up within next 30 days	Some S.G.s C-2 but no additional S.G. ~~are~~ is C-3	Perform action for C-2 result of second sample other	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

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A.1

5.0 Admin Controls

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CTS

6.0 ADMINISTRATIVE CONTROLS

5.5.9 | 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.

CTS

6.0 ADMINISTRATIVE CONTROLS

5.5.10 6.5.10 Ventilation Filter Testing Program

Handling Area

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.1

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%

CRV

e. Demonstrate that the heaters for ~~each of~~ the ~~ventilation~~ system~~s~~ dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989: A.1

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

SR 3.0.2 and SR 3.0.3 A.1

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.

5.5.11 6.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

1. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

   a) API gravity or an absolute specific gravity,

   b) Kinematic viscosity, and

   c) Water and sediment content.

2. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program. A.11

5.5.12 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 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). A.1

6.0 ADMINISTRATIVE CONTROLS

5.5.13 6.5.13 ~~Reserved~~

< ADD Safety Functions Determination Program (SFDP) as presented in ITS > M.4

5.5.14 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)~~. A.1

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 $\geq$ 10 psig.

The Surveillance interval extensions of ~~LCO 4~~ SR 3.0.2 are not applicable to the Containment Leak Rate Testing Program requirements. A.1

The provisions of ~~LCO 4~~ SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements. A.1

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October 31, 1996

5.5.15 | 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.

6.0 ADMINISTRATIVE CONTROLS

CTS 5 ~~6~~.6 REPORTING REQUIREMENTS

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

5 ~~6~~.6.1 Occupational Radiation Exposure Report

INSERT

~~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.~~ A.12

5 ~~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 (ODCM) and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5 ~~6~~.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering operation of the plant in the previous year shall be submitted prior to May 1 of each year ~~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 (ODCM) and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.~~ A.12

5 ~~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. A.13

SECTION 5.0

INSERT 1

This report shall include a tabulation on an annual basis of the number of stations, utility and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent greater than 100 mrem and the associated deep dose equivalent (reported in person-rem) 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 assignments to various duty functions may be estimated based on pocket ionization chamber, 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 deep dose equivalent external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

6.0 ADMINISTRATIVE CONTROLS

5.6.5 ~~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:

3.2.4 ~~3.1.1~~ ASI Limits.
3.1.6 3.10.5 Regulating Rod Group Position ~~Insertion~~ Limits
3.2.1 ~~3.23.1~~ Linear Heat Rate (LHR) Limits
3.2.2 ~~3.23.2~~ Radial Peaking Factor Limits

A.1

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~~ 3.2.4, ~~3.10.1~~, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1 A.8

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~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

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~~ 3.2.4, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs ~~3.10.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

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~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

   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.

5.5.5 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~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

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~~ 3.2.4, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

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~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

10. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs ~~3.1.1~~ 3.2.4, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

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~~ A.1

Post

5.6.6 6.6.~~7~~ 6 Accident Monitoring Instrument Report Post

When a report is required by ~~Condition 3.17/4.7c~~ LCO 3.3.7, "Accident Monitoring Instrumentation," a report shall be submitted within the following ~~30~~ 14 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. A.1 M.5

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6.0 ADMINISTRATIVE CONTROLS

5.6.7 6.6.~~6~~7 Containment Structural Integrity Surveillance Report

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

(A.9) (M.3)

5.6.8 6.6.~~7~~8 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.

6.0 ADMINISTRATIVE CONTROLS

6.7 HIGH RADIATION AREA

5.7.1 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.

Radiation Safety

A.1

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.

A.1

5.7.2 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.

5 5 5

A.1 Radiation Safety

5.7.3 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.

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involve no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 6.1.2, 6.2.2a and 6.2.2b use the terminology "above COLD SHUTDOWN." In the proposed ITS, this corresponds to MODES 1, 2, 3, and 4. As discussed in Chapter 1.0, the CTS COLD SHUTDOWN is essentially equivalent to the ITS MODE 5 (CTS 210 F vs. ITS 200 F). Therefore, "above COLD SHUTDOWN" in the CTS equates to MODES 1, 2, 3, and 4 in the proposed ITS. This change is considered to be an administrative change to adopt the terminology of the ISTS.

A.3 CTS 6.2.2a uses the phrases "assigned to each reactor containing fuel," and "assigned for each control room." The Palisades Nuclear Plant has only one reactor and one control room. Therefore, the wording in ITS 5.2.2 is being modified to state "assigned when fuel is in the reactor," and "assigned when the reactor is operating" to more accurately reflect the Palisades plant specific design. This change is considered to be an administrative change since no technical requirements have changed.

A.4 CTS 6.2.2b, 6.2.2g, and 6.5.4d use the term "unit" when discussing the reactor. The typical term used in the remainder of the CTS is "plant." Therefore, the term "plant" will be used in the proposed ITS 5.2.2. This is an administrative change to reflect the typical Palisades Nuclear Plant terminology.

A.5 CTS 6.4.1 requires that written procedures shall be established, implemented, and maintained for the activities listed. In this list, the CTS contains item b., "Refueling operations, and item c., "Surveillance and test activities of safety-related activities." These items are included in the procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978 which is referenced in CTS 6.4.1a and included in the proposed ITS 5.4.1a. Therefore, since these procedures are already required by the reference to Regulatory Guide 1.33, Revision 2, February 1978, they are not included in the proposed ITS. This change is an administrative change since no requirements have changed. This change maintains consistency with NUREG-1432.

A.6 CTS 6.4.1 requires that written procedures shall be established, implemented, and maintained for the activities listed. In this list, the CTS contains item f., "Site Security Plan implementation" and item g.,"Site Emergency Plan implementation.." These items were recommended to be removed from the Technical Specifications in NRC Generic Letter 93-07 since they are duplicative of regulations contained in the Code of Federal Regulations part 50 and 73. This change is considered to be an administrative change since these requirements must still be met as required by the Code of Federal Regulations. This change maintains consistency with NUREG-1432.

A.7 CTS 6.5.7 is entitled "Inservice Inspection and Testing Program." In the proposed ITS 5.5.7, the title is changed to the "Inservice Testing Program." This change is considered to be an administrative change since the requirements of the program are unchanged. This change maintains consistency with NUREG-1432.

A.8 CTS 6.6.5b.1 lists, among referenced LCOs, "3.10.1." That item is unnecessary and has been deleted. Neither CTS 3.10.1, nor its ITS replacement reference the COLR.

A.9 CTS 6.6.8, "Containment Structural Integrity Surveillance Report" requires that a report be submitted to the NRC covering Prestressing, Anchorage, and Liner and Penetration tests. Proposed ITS 5.6.7, "Containment Structural Integrity Surveillance Report" also requires that a report be submitted to the NRC but only specifies the Prestressing and Anchorage tests be included. Reference to the Liner and Penetration tests have been deleted since the requirement for these tests was removed from the technical specifications by Amendment 109 dated October 28, 1987. Initially, the Liner and Penetration tests were included in the CTS since they were relative new designs and a surveillance program was established to assure the affected components would maintain their functional integrity. Based on test data, it was concluded that the liner plates and penetration assemblies were performing as predicted. Therefore, the CTS was amended and the surveillance program terminated. As such, it is no longer necessary to reference these tests in ITS 5.6.7.

A.10 CTS 4.5.6, "Dome Delamination Surveillance" has been modified to include reference to ITS 5.6.7, "Containment Structural Integrity Surveillance Report." The intent of this change is to clarify the reporting requirements associated with the dome delamination inspection. As stated in CTS 4.5.6, a dome delamination inspection shall be performed within 90 days following corrective retentioning of dome tendons and the results of the inspection reported to the NRC. ITS 5.6.7 requires that a report of the dome delamination test be submitted to the NRC within 90 days after completion of the test. The proposed change is considered administrative in nature since no additional restriction are imposed on plant operation. Inclusion of the dome delamination reporting requirements in the Containment Structural Integrity Surveillance Report is discussed in Discussion of Change M.3 to this Section.

A.11 CTS 6.5.8, "Steam Generator Tube Surveillance Program," and CTS 6.5.11, "Fuel Oil Testing Program," are revised to provide statements of applicability for SR 3.0.2 and for SR 3.0.2 and SR 3.0.3, respectively. These statements provide clarity and ensure consistent application of these requirements for the Programs referenced by ITS SRs. This change is consistent with NUREG-1432 as modified by TSTF-118.

A.12 CTS 6.6.1, "Occupational Radiation Exposure Report," and CTS 6.6.3, "Radioactive Effluent Release Report," are revised to incorporate language related to revisions to 10 CFR Part 20, and 10 CFR 50.36a. These changes are administrative since there are not actual changes in the application of the requirements. This change is consistent with NUREG-1432 as modified by TSTF-152.

A.13 CTS 6.6.4, "Monthly Operating Report," is revised to omit the words "to arrive" since the Palisades Nuclear Plant has no control of the document once it is mailed. Further, this is inconsistent with typical NRC submittal requirements. This change is considered administrative since it has no effect on plant operations and impacts only the submittal of after-the-fact information. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 6.4.1 requires that written procedures be established, implemented, and maintained for the listed activities. Proposed ITS 5.4.1 contains the same wording. However, proposed ITS 5.4.1.b is not in the CTS and is being added. Proposed ITS 5.4.1.b states "The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33." Since this item is not included in the CTS it is considered to be a more restrictive change. This change maintains consistency with NUREG-1432.

M.2 CTS 6.5.3 describes the Post Accident Sampling Program. It states in part "..and which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents,...." In the proposed ITS, the reference is to "radioactive gases" rather than just radioactive iodines. Because the use of the term "gases" is broader than "iodines" for the sampling and analyzing requirements, this is considered to be a more restrictive change. This change is consistent with NUREG-1432.

M.3 The CTS does not contain a program for Containment Tendon Testing. CTS Sections 4.5.4 and 4.5.5 do address tendon testing and these requirements have been replaced with a program. CTS 4.5.6 contains requirements for containment dome delamination inspection. These dome delamination inspection requirements have been added to the ISTS program requirements. Since the program addresses structural components other than tendons, the program has been titled "Containment Structural Integrity Surveillance Program."

M.4 The CTS does not contain a Safety Functions Determination Program. Proposed ITS 5.5.13 includes this program. This program is added to work in conjunction with the proposed ITS in identifying any loss of safety function which might exist. Because the CTS did not contain this program, and its implementation requires additional evaluations to identify a loss of safety function than what is required in the CTS, this change is considered to be a more restrictive change. This change maintains consistency with NUREG-1432.

M.5 CTS 6.6.7 contains the reporting requirements for specific accident monitoring instrument channels that are not restored to an Operable status within the required Completion Time. CTS 6.6.7 requires that a report be submitted within 30 days. Proposed ITS 5.6.6 also contains reporting requirements for specific accident monitoring instrument channels that are not restored to an Operable status within the required Completion Time. However, the ITS requires that a report be submitted within 14 days. As such, the proposed change imposes an additional restriction on plant operations since the time period allowed to submit the report has been shortened from 30 days to 14 days. This change has been proposed to establish consistency with NUREG-1432 and is deemed acceptable since it only involves a change to administrative requirements and does not alter the way in which the plant is operated.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

LA.1 CTS Specification 4.5.4, Surveillance for Prestressing System (page 4-21a) and 4.5.5, End Anchorage Concrete Surveillance (page 4-21c) were replaced by proposed ITS Specification 5.5.5, the Containment Structural Integrity Surveillance Program. The proposed specification emulates the ISTS treatment of containment structural integrity surveillance requirements. The details associated with containment tendon inspections have been removed from the technical specification and reference has been included in ITS 5.5.5 to ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL which establishes the applicable test methods, acceptance criteria and testing frequencies. Removal of these details is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL is specified in 10 CFR 50.55a. Thus, this change eliminates duplication of federal regulations and can be made without an impact on public health and safety. Removal of these details from the CTS and the incorporation of a containment tendon surveillance program in Section 5.0 of the ITS is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive" changes made to this chapter.

RELOCATED (R)

There were no "Relocated" changes made to this chapter.

ATTACHMENT 4

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES

The Palisades Nuclear Plant is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording which does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1432. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, Palisades Nuclear Plant staff has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1432. During the development of NUREG-1432, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in margin of safety?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

MORE RESTRICTIVE CHANGES

The Palisades Nuclear Plant is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements which currently do not exist.

These changes may include additional requirements that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1432.

In accordance with the criteria set forth in 10 CFR 50.92, the Palisades Nuclear Plant has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in margin of safety?

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS

The Palisades Nuclear Plant is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve moving details (engineering, procedural, etc.) out of the Technical Specifications and into a licensee controlled document. This information may be moved to the ITS Bases, FSAR, plant procedures or other programs controlled by the licensee. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1432 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, Palisades Nuclear Plant staff has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The proposed changes move details from the Technical Specifications to a licensee controlled document. The removal of details from the Technical Specifications is not assumed to be an initiator of any analyzed event. The proposed changes do not reduce the functional requirement or alter the intent of any specification. As such, the consequences of an accident remain unchanged. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed changes move detail from the Technical Specifications to a licensee controlled document. The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. There are no design changes or equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being proposed in the plant operational limits as a result of this change. The proposed changes remove details from the Technical Specifications and place them under licensee control. Removal of these details is acceptable since this information is not directly pertinent to the actual requirement and does not alter the intent of the requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to licensee controlled document without a significant impact on safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES

There were no "Less Restrictive Changes" made in Chapter 5.

ATTACHMENT 5

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

MARKUP OF NUREG-1432

TECHNICAL SPECIFICATIONS

<CTS>

6.0 5.0 ADMINISTRATIVE CONTROLS

6.1 5.1 Responsibility

6.1.1 5.1.1 The [Plant Superintendent] shall be responsible for overall unit plant operation and shall delegate in writing the succession to this responsibility during his absence. (1)

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. (1)

6.1.2 5.1.2 The [Shift Supervisor (SS)] shall plant be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. (1) During any absence of the [SS] from the control room while the plant unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function. (1)

6.0 5.0 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant. Palisades (4)

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the [FSAR]; (1) (5)

<INSERT> →

b. The [Plant Superintendent] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; (1)

c. The [a specified corporate executive ~~position~~] 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; and (1) (5)

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Plant Staff

The unit staff organization shall include the following: (4)

a. A non-licensed operator shall be assigned when fuel is in the to each reactor containing fuel and an additional non-licensed operator (6)

(continued)

SECTION 5.0

INSERT

These requirements and the plant specific equivalent of those titles referred to in these Technical Specifications shall be documented in the FSAR.

5.2 Organization

5.2.2 PLANT ~~Unit~~ Staff (continued) (4)

WHEN THE

shall be assigned ~~for each control room from which a~~ reactor (6)
is operating in MODES 1, 2, 3, or 4.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.] (3)

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 PLANT ~~unit~~ is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. (4)

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.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 minimum requirements.

d. A [~~Health Physics 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. (4) (1)

e. Administrative procedures shall be developed and implemented to limit the working hours of PLANT ~~unit~~ staff who perform safety related functions (e.g., licensed SROs, licensed ROs, ~~health physicists~~ radiation Safety Personnel, auxiliary operators, and key maintenance personnel). (4) (1)

<INSERT> (5)

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

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

(continued)

SECTION 5.0

INSERT

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

5.2 Organization

5.2.2 PLANT ~~Unit~~ Staff (continued) 4

[

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.

Any deviation from the above 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. 1 5

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. 5

]

~~OR~~

~~[The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).]~~ 3

f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license. 5 The individual holding the SRO license shall be responsible for directing the activities of the licensed operators; 7

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 plant ~~unit~~. ~~In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~ If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed. 4 8 9

5.0 ADMINISTRATIVE CONTROLS

5.3 Plant ~~Unit~~ Staff Qualifications

~~[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]~~

5.3.1 Each member of the plant ~~unit~~ staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff].~~ ANSI N18.1-1971 For comparable positions 1

~~The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ 10

<INSERT> 10

SECTION 5.0

INSERT

5.3.2 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.

5.3.3 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 accident. The Shift Engineer shall hold a Senior Reactor Operator license.

5.3.4 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.

CTS

6.0 # 5.0 ADMINISTRATIVE CONTROLS

6.4 ## 5.4 Procedures

6.4.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in {Generic Letter 82-33}; ① ①

~~c. Quality assurance for effluent and environmental monitoring;~~ ⑪

(c) ~~d.~~ Fire Protection Program implementation; and

d) ~~e.~~ All programs specified in Specification 5.5.

~~f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.~~ ③

~~Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.~~

CTS

5.0 ADMINISTRATIVE CONTROLS

6.5 5.5 Programs and Manuals

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

6.5.1 5.5.1 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 the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.6.2.] and Specification [5.6.3]. 1

Licensee initiated changes to the ODCM: 5

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

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 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;

b. Shall become effective after the approval of the [Plant Superintendent]; and 1 5

c. 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 in the ODCM was made. Each change shall be identified by markings in the margin of

(continued)

CTS

5.5 Programs ~~and~~ Manuals

6.5.1 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

6.5.2 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]. The program shall include the following: 1

Containment

containment sump suction piping

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less;

<Insert> 12

6.5.3 5.5.3 Post Accident Sampling

accurately determine the airborne iodine concentration in vital areas and which will ensure the capability to 13

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

6.5.4 5.5.4 Radioactive Effluent Controls Program

This program conforms to 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

(continued)

SECTION 5.0

INSERT

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; and

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).

5.5 Programs and Manuals

6.5.4 5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability 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 times the value in 10 CFR 20, Appendix B, Table 2, Column 2; (5)

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations 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. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~ (14)

~~f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;~~

e. ~~g.~~ Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 times the value listed in 10 CFR 20, Appendix B, Table 2, Column 1; (5)

(continued)

5.5 Programs and Manuals

6.5.4 5.5.4 Radioactive Effluent Controls Program (continued)

f ~~k~~. 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 ~~l~~. 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 > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

h ~~m~~. Limitations on the annual dose 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.

~~5.5.5 Component Cyclic or Transient Limit~~

~~This program provides controls to track the FSAR Section [ ] cyclic and transient occurrences to ensure that components are maintained within the design limits.~~ 15

Containment Structural Integrity Surveillance Program 23

new 5.5.~~6~~ 5 ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The ~~Tendon~~ Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1989]~~ ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL 1 2

4.5.6 <INSERT>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the ~~Tendon~~ CONTAINMENT STRUCTURAL INTEGRITY Surveillance Program inspection frequencies. 23

6.5.6 5.5.~~7~~ 6 Primary ~~Reactor~~ Coolant Pump Flywheel Inspection Program 4

~~This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~ 16

Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

The provisions of SR 3.0.2 are applicable to the Primary Coolant Pump Flywheel Inspection Program

(continued)

SECTION 5.0

INSERT

If, as a result of a tendon inspection, 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, 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 in accordance with Specification 5.6.7, "Containment Structural Integrity Surveillance Report."

5.5 Programs and Manuals (continued)

6.5.7 5.5.~~8~~ 7 Inservice Testing Program

This program provides controls for inservice 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 as follows: (B&PV Code) 5

B&PV Code ~~ASME Boiler and Pressure Vessel Code and applicable Addenda~~ terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	≤ ~~At least once per~~ 7 days
Monthly	≤ ~~At least once per~~ 31 days
Quarterly or every 3 months	≤ ~~At least once per~~ 92 days
Semiannually or every 6 months	≤ ~~At least once per~~ 184 days
Every 9 months	≤ ~~At least once per~~ 276 days
Yearly or annually	≤ ~~At least once per~~ 366 days
Biennially or every 2 years	≤ ~~At least once per~~ 731 days

5

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the B&PV ~~ASME Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any ~~TS~~. 5
TECHNICAL SPECIFICATION 2

6.5.8 5.5.~~9~~ 8 Steam Generator (SG) Tube Surveillance Program

[~~Reviewer's Note: The Licensees current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~] 1

<INSERT SG Tube Surveillance Program from CTS> 5 14

(continued)

SECTION 5.0

INSERT

5.5.8 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 5.5.8-1. 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;

SECTION 5.0

INSERT (continued)

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.

3. The tubes selected as the second and third samples (if required by Table 5.5.8-1) 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.

SECTION 5.0

INSERT (continued)

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;

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.8-1 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 5.5.8.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 5.5.8-1 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 LCO 3.4.13;

    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;

SECTION 5.0

INSERT (continued)

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;

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 describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.8.b.3, above;

SECTION 5.0

INSERT (continued)

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 5.5.8-1.

This provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program. 24

SECTION 5.0

<u>INSERT TABLE</u>

TABLE 5.5.8-1

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION[1]		2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Result	Action Required	Result	Action Required	Result	Action Required
C-1	None	N/A	N/A	N/A	N/A
C-2	Plug defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG.	C-2	Plug defective tubes and inspect additional 4S tubes in this SG.	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
		All other SGs are C-1	None	N/A	N/A
		Some SGs C-2 but no other SG is C-3	Perform action for C-2 result of second sample	N/A	N/A
		Other SG is C-3	Inspect all tubes each SG and plug defective tubes	N/A	N/A

NOTES: 1 The minimum sample size for the first sample inspection is S tubes per SG where S=(6/n)%, where n is the number of steam generators inspected during an inspection.

5.5 Programs and Manuals (continued)

CTS

6.5.9 **5.5.~~10~~ 9** Secondary Water Chemistry Program 2

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 5

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; (HYPHEN) 5

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 the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

6.5.10 **5.5.~~11~~ 10** Ventilation Filter Testing Program (~~VFTP~~)

Control Room Ventilation (CRV) and Fuel Handling Area Ventilation

A program shall be established to implement the following required testing of ~~Engineered Safety Feature (ESF) filter/ventilation~~ systems at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989~~, and AG-1~~] at the system flowrate and tolerances specified below* ~~[± 10%]~~. 4 1 1.52 Revision 2 1 4

a. **Demonstrate** for each of the ~~ESF~~ 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 Fuel Handling Area Ventilation System when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME 4 1 1

* 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. 17

(continued)

CTS

5.5 Programs and Manuals

6.5.10 5.5.11 10 Ventilation Filter Testing Program (VFTP) (continued)

N510-1989, at the system flowrate specified as follows [± 10%]: 2

ESF 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%

1

b. Demonstrate for each of the ESF ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [± 10%]: 4 1 2

For the CRV and < 1.00% for the Fuel Handling Area Ventilation System

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

1

c. Demonstrate for each of the ESF ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], 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 greater than or equal to the relative humidity specified as follows: 4 1 1 1

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

1

(continued)

5.5 Programs and Manuals

CTS 6.5.10

5.5.~~11~~ 10 Ventilation Filter Testing Program (VFTP) (continued)

[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters.] (2)

ventilation

d. For each of the ~~ESF~~ 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 ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ ~~at the system flowrate specified as follows [± 10%]~~: (4) (1) (2)

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

(4) (1)

e. Demonstrate that the heaters for ~~each of~~ the ~~ESF~~ CRV system~~s~~ dissipates the following specified value ~~[± 10%]~~ when tested in accordance with ~~[ASME N510-1989]~~: (1) (2)

~~ESF~~ Ventilation System	Wattage
VHX-26A and VHX-26B	15 KW

(4) (1)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

~~5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program~~

~~This program provides control for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The~~ (13)

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the [Waste Gas Holdup System] and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

(18)

(continued)

5.5 Programs and Manuals (continued)

CTS
6.5.11

5.5.13 1 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has: 5

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. A clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

<INSERT>

24

5.5.14 2 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:

   A change in the TS incorporated in the license; or

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

(continued)

SECTION 5.0

INSERT

Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

1. API gravity or an absolute specific gravity,
2. Kinematic viscosity, and
3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program. 24

5.5 Programs and Manuals

CTS

5.5.12 5.5.1~~0~~2 Technical Specifications (TS) Bases Control Program (continued)

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 5.5.1~~4b~~ 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). 2

new 5.5.1~~5~~3 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals

CTS

new 5.5.18 3 Safety Functions Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

6.5.14 5.5.14 <Insert Containment Leak Rate Testing Program> 19

6.5.15 5.5.15 <INSERT PROCESS CONTROL PROGRAM> 5

SECTION 5.0

INSERT 1

5.5.14 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.

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

Local leak rate tests, other than Personnel Airlock doors between the seals tests, shall be performed at ≥ 55 psig.

Local leak rate tests for checking airlock doors seals within 72 hours of each door opening shall be performed as follows:

a. A between the seals test shall be performed on the Personnel Airlock at ≥ 10 psig.

b. A full pressure test shall be performed on the Emergency Escape Airlock at ≥ 55 psig. A seal contact check shall be performed on the Emergency Escape Airlock following each full pressure test. Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

SECTION 5.0

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. The leakage for a Personnel airlock door seal test shall not exceed $0.023\ L_a$.

c. An acceptable Emergency Escape Airlock door seal contact check consists of a verification of continuous contact between the seals and the sealing surfaces.

Containment OPERABILITY is equivalent to "Containment Integrity" for the purposes of the air lock testing requirements in 10 CFR 50, Appendix J.

The provisions of SR 3.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

SECTION 5.0

INSERT 2

5.5.15 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 waster product to existing requirements of Federal, State, or other applicable regulations.

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

5.0 ADMINISTRATIVE CONTROLS

CTS

6.6 5.6 Reporting Requirements

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

6.6.1 5.6.1 Occupational Radiation Exposure Report

-----------------------------------NOTE-----------------------------------
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~ (3)

INSERT

~~A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.~~ (25) ~~[The initial report shall be submitted by April 30 of the year following initial criticality.]~~ (3)

6.6.2 5.6.2 Annual Radiological Environmental Operating Report (5)

-----------------------------------NOTE-----------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. (3)

The ~~Annual~~ Radiological Environmental Operating Report covering (5) **the operation of the** unit plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses 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

(continued)

SECTION 5.0

INSERT 1

This report shall include a tabulation on an annual basis of the number of stations, utility and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent greater than 100 mrem and the associated deep dose equivalent (reported in person-rem) 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 assignments to various duty functions may be estimated based on pocket ionization chamber, 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 deep dose equivalent external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6 Reporting Requirements

CTS 6.6.2

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

~~The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.~~ (5)

6.6.3 5.6.3 Radioactive Effluent Release Report

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~ (3)

prior to May 1 of each year

plant in the previous year

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The (25) report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant unit. The material provided shall be consistent with the (4) objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer (1)

(continued)

5.6 Reporting Requirements

CTS 6.6.4

5.6.4 Monthly Operating Reports (continued)

[power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. 1

6.6.5

5.6.5 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:

LCO 3.1.6 Regulating Rod Group Position Limits
LCO 3.2.1 Linear Heat Rate Limits
LCO 3.2.2 Radial Peaking Factor Limits
LCO 3.2.4 ASI Limits

~~[The individual specifications that address core operating limits must be referenced here.]~~ 1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in THE LATEST REVISION OF the following documents: 5

<INSERT>

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

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 (ECCS) limits, nuclear limits such as SDM, 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.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic~~ 20

(continued)

SECTION 5.0

INSERT

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs" and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EXEM PWR Large Break LOCA Model as defined by:
   (LCOs 3.1.6, 3.2.1, & 3.2.2)

   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.

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<u>INSERT</u> (continued)

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.

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 3.1.6, 3.2.1, & 3.2.2)

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 3.2.1, 3.2.2, & 3.2.4)

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 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

10. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.2.1, 3.2.2, & 3.2.4)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the return fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

20

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology. (20)

5.6.7 EDG Failures Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any non valid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement. (21)

CTS 6.6.7 5.6.8 6 Post Accident Monitoring PAM Report

When a report is required by Condition B or G of LCO 3.3.[11] 7, 2 "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 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 of the Function to OPERABLE status. (1) (2)

6.6.8 5.6.9 7 Containment Structural Integrity Tendon Surveillance Report

9 Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 9 (22)

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

Dome Delamination tests

(continued)

CTS

5.6 Reporting Requirements

6.6.8 **5.6.~~9~~7** Tendon Surveillance Report (continued)

~~30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~ 1

6.6.9 **5.6.~~10~~8** Steam Generator Tube Inspector Report

<INSERT>

~~Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~ 1

~~Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~ 1

SECTION 5.0

INSERT

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.

5.0 ADMINISTRATIVE CONTROLS

{5.7 High Radiation Area} ①

6.7.1 5.7.1 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 < ⑤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. ① ⑤ ⑤

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following: ①

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 Protection Manager} in the~~ RWP. ①

6.7.2 5.7.2 In addition to the requirements of Specification 5.7.1, except as allowed by 5.7.3 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 ~~Foreman~~ Supervisor on duty or ~~health physics supervision~~ Radiation Safety Supervision. Doors shall remain locked except during periods of access by personnel

(continued)

CTS

[5.7 High Radiation Area]

6.7.2 5.7.2 (continued)

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 5.7.3 For individual high radiation areas with radiation levels of 5 $\geq$ 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.

ATTACHMENT 6

PALISADES NUCLEAR PLANT

CHAPTER 5.0, ADMINISTRATIVE CONTROLS

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1432

Change **Discussion**

Note: This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.

1. The brackets have been removed and the proper plant specific information or value has been provided.

2. Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specification. The Administrative Controls section of the Palisades Technical Specifications has been recently amended to closely emulate NUREG-1432. In some instances the wording and requirements differ from those in NUREG-1432. In order to limit the impact of conversion to improved Technical Specifications, revision of the subject wording has not be proposed.

6. CTS 6.2.2a uses the phrases "assigned to each reactor containing fuel," and "assigned for each control room." The Palisades Nuclear Plant has only one reactor and one control room. Therefore, the wording in ITS 5.2.2 is being modified to state "assigned when fuel is in the reactor," and "assigned when the reactor is operating" to more accurately reflect the Palisades plant specific design.

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7. CTS 6.2.2f requires the operations manager or the assistant operations manager to hold an SRO license. In addition, it states "The individual holding the SRO license shall be responsible for directing the activities of the licensed operators." This statement is added to proposed ITS 5.2.2f to provide clarification on who directs the activities of the licensed operators in the situation where the operations manager or the assistant operations manager does not hold an SRO license.

8. Proposed ITS 5.2.2g states in the last sentence "In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift." This statement is not included in the proposed Palisades ITS. The Administrative Controls Chapter of the Palisades Technical Specifications was recently changed in Amendment 174 dated October 31, 1996 to be consistent with the format and presentation presented in NUREG-1432. The NUREG-1432 statement was not included in the subject amendment and Palisades feels that the existing wording in the CTS, including the qualifications requirements in proposed ITS 5.3.3 adequately describes the STA's role and qualification requirements. This is a plant specific change based on the Palisades Nuclear Plant CTS.

9. In proposed ITS 5.2.2g the following statement is added: "If either SRO on shift satisfies the Shift Engineer qualification requirements, then the STA does not need to be stationed." In CTS 6.3.3 and proposed ITS 5.3.3, the Shift Engineer is required to have a bachelor's degree in a scientific or engineering discipline and have specific training in plant design, operations, and response and analysis of the plant for transients and accidents. Because the training and qualifications of the Shift Engineer encompasses that of the STA, an SRO on shift who meets the Shift Engineer qualifications provides the equivalent level of qualifications as the STA, and therefore the STA does not need to be stationed. This is a plant specific change based on the Palisades Nuclear Plant CTS.

10. NUREG 1432 Item 5.3.1 in the second sentence discusses "The staff not covered by ....." The proposed ITS deletes this second sentence and replaces it with three separate subsections discussing 1) the radiation safety manager qualifications; 2) the Shift Technical Advisor and Shift Engineer qualifications; and 3)the qualification requirements for plant staff who perform 50.59 reviews. This change is consistent with the "Reviewer's Note" in Section 5.3 of NUREG-1432. This is a plant specific change based on the Palisades Nuclear Plant CTS.

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11. NUREG-1432 contains Item 5.4.1c which requires that written procedures shall be established, implemented, and maintained for "Quality assurance for effluent and environmental monitoring." The Palisades Nuclear Plant CTS does not contain this requirement. The Administrative Controls Chapter of the Palisades Nuclear Plant was recently changed in Amendment 174, dated October 31, 1996 to be consistent with the format and presentation of NUREG-1432. The subject NUREG-1432 statement was not included in this amendment and Palisades feels that the requirements for effluent and environmental monitoring are adequately prescribed in the ODCM. Therefore, the subject wording is not included in the proposed Palisades ITS. This is a plant specific change based on the Palisades Nuclear Plant CTS.

12. The proposed ITS 5.5.2 contains requirements for the program which addresses Primary Coolant Sources Outside Containment. In CTS 6.5.2 which contains the same program, three additional aspects of the program are listed to address plant specific applications. Part "c" addresses the testing requirements for the shutdown cooling system that is outside the containment. Part "d" discusses the testing requirements for piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps. Part "e" discusses the leakage requirements from the recirculation heat removal systems' components. These items will be included in proposed ITS 5.5.2. These changes are plant specific changes to reflect the CTS and plant specific requirements.

13. The proposed ITS 5.5.3, Post Accident Sampling, is a which provides controls to ensure the capability to obtain and analyze samples under post accident conditions. An additional requirement is included in the proposed ITS which is taken from CTS 6.5.3, Post Accident Sampling, to ensure the capability to accurately determine the airborne iodine concentration in vital areas. This is a plant specific change to retain the requirements contained in the CTS Post Accident Sampling program.

14. NUREG-1432 5.5.4 contains the Radioactive Effluent Controls Program. Items "e" and "f" of this program are not included in the proposed ITS as they were not part of the Palisades Radioactive Effluent Controls Program contained in CTS 6.5.4. This change is a plant specific change to reflect the Palisades CTS. The Palisades Administrative Controls Chapter was recently changed in Amendment 174 dated October 31, 1996 to match the format and presentation of NUREG-1432. The CTS does not contain NUREG-1432 items "e" and "f" and therefore, they are not included in the proposed ITS.

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23.	NUREG-1432 5.5.6 specifies the requirements for the Tendon Surveillance Program. The proposed ITS revises this title to the Containment Structural Integrity Surveillance Program and replaces the plant specific requirements which currently exists in CTS 4.5.4 and 4.5.5. This program and the associated report (ITS 5.6.7), include requirements relating to dome delamination in addition to tendon testing. The program and report names were changed accordingly. This change is acceptable because the NUREG-1432 5.5.6 is in brackets to indicate that the plant specific information is to be provided if the report is applicable.
24.	NUREG-1432 5.5.9 and 5.5.13 are revised to incorporate TSTF-118 which provides consistent application of SR 3.0.2 and SR 3.0.3 to the Programs referenced by ITS SRs.
25.	NUREG-1432 5.6.1 and 5.6.3 are revised to incorporate TSTF-152 which reflects previous revisions to 10 CFR Part 20 and 10 CFR 50.36a.

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15. NUREG-1432 Section 5.5.5 contains the program requirements for the Component Cyclic or Transient Limits. The Palisades CTS does not contain an equivalent program and therefore, it is not included in the proposed ITS.

16. NUREG-1432 Section 5.5.7 contains the program requirements for the Reactor Coolant Pump Flywheel Inspection Program. The NUREG-1432 program references Regulatory Guide 1.14. The proposed ITS 5.5.6, Primary Coolant Pump Flywheel Inspection Program, is based on CTS 6.5.6. The proposed ITS program requires "Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years." This wording is a direct transfer of the CTS requirements in 6.5.6, as modified in the Palisades Technical Specification Change Request submitted October 1, 1997. These changes are plant specific changes to reflect the Palisades CTS requirements.

17. NUREG-1432 Section 5.5.11 contains the requirements for the Ventilation Filter Testing Program (VFTP). In the proposed ITS 5.5.10 a footnote is added to the first paragraph to modify the testing frequencies. The proposed footnote states: "Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal- such as refueling-testing may be delayed until the completion of the plant operation or up to 1500 hours of filter operation, whichever occurs first." This note is used in CTS 6.5.10 which contains the requirements for the VFTP. This note is added to accommodate normal operational occurrences which otherwise might impose an unnecessary burden on plant operations. This change is a plant specific change to reflect the Palisades CTS.

18. NUREG-1432 Section 5.5.12 contains requirements for the Explosive Gas and Storage Tank Radioactivity Monitoring Program. The Palisades CTS does not contain these requirements and therefore they are not included in the proposed ITS.

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19. Proposed ITS 5.5.14 contains the Containment Leak Rate Testing Program. This program was added to NUREG-1432 as part of TSTF-52 for the implementation of OPTION B to 10 CFR 50 Appendix J. This program has already been adopted in the CTS in section 6.5.14. The CTS program described in 6.5.14 is included as the proposed ITS 5.5.14, Containment Leak Rate Testing Program, with appropriate information from CTS 4.5.2 concerning test methods and acceptance criteria.

    The proposed ITS Containment Leak Rate Testing Program includes specific acceptance criteria for each airlock. These criteria and the associated exemption to 10 CFR 50 Appendix J were approved as Amendment 177 to the Palisades Technical Specifications on September 30, 1997.

    An additional paragraph was included to assure correct application of those 10 CFR 50 Appendix J testing requirements (e.g., III.D.2.(b)(ii)) which are applicable "when containment integrity is required by the plant's Technical Specifications."

20. NUREG-1432 5.6.6 contains the requirements for the Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR). The Palisades Nuclear Plant does not currently utilize a PTLR and does not propose to include one in the proposed ITS.

21. NUREG-1432 5.6.7 contains the requirements for an EDG failures report. The Palisades Nuclear Plant CTS does not have an EDG failures report and does not propose to include one in the proposed ITS. This report is shown in brackets in NUREG-1432 to indicate that plants which utilize this report should put in the plant specific information. Therefore, it is not applicable to the Palisades ITS.

22. NUREG-1432 5.6.9 specifies the requirements for the Tendon Surveillance Report. The proposed ITS revises this title to the Containment Structural Integrity Report and includes the plant specific information which currently exists in CTS 6.6.8. This report and the associated program (ITS 5.5.5), include requirements relating to dome delamination in addition to tendon testing. The program and report names were changed accordingly. This change is acceptable because the NUREG-1432 5.6.9 is in brackets to indicate that the plant specific information is to be provided if the report is applicable.