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U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555-0001

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Administrative Matters Relating to Outstanding
Requests for Technical Specification Changes

Dear Sir:

The purpose of this letter is to transmit revised pages for Technical Specification (TS) changes previously proposed as well as to docket other administrative matters relating to currently proposed TS changes. TS Amendment 205 changed the TS in its entirety thus necessitating the revision of TS pages submitted prior to Amendment 205. Please note that the attached pages include complete sections of the TS since we plan to replace sections as a whole. This has administrative benefits including the need to track pages by amendment number. The following identify proposed Technical Specification changes by reference, and the scope of this transmittal:

1. Reference 1 proposed a change to the administrative section of the TS regarding operations qualifications. Entergy requests that this proposed amendment be withdrawn since Amendment 205 eliminated the need for the change.

2. Reference 2 proposed a one-time change to the TS to extend the Type A integrated leak rate test until December 1, 2005. Attachment 1 contains TS section 5.0 with the proposed change.

3. Reference 3 proposed a change to extend the duration of the testing interval for the Fuel Storage Building Charcoal. Attachment 2 contains TS section 5.0 with the proposed change.

4. Reference 4 proposed a change to the TS to address Generic Letter 99-02 concerns for

A001

charcoal testing. Entergy requests that this proposed amendment be withdrawn since additional design information for charcoal systems has been developed that requires reconsideration of this submittal. The proposed TS change will be resubmitted when it is revised.

5. Reference 5 proposed a change to the TS to reflect a planned plant modification that would allow a change to the isolation valves on the main feedwater lines credited for a postulated main steam line break event. Attachment 3 contains TS section 3.7.3 with the proposed change. The bases pages associated with this TS change will be distributed in accordance with the TS Bases Control Program.

6. Reference 6 proposed a one time change to the TS to allow on line replacement of station batteries. Attachment 4 contains TS section 3.8.4 with the proposed change.

7. Reference 7 proposed a one time change to the TS to allow on line refurbishment of emergency diesel generator fuel oil storage tanks. Attachment 5 contains TS section 3.8.1 and 3.8.3 with the proposed change.

Indian Point 3 is making no commitments in this letter.

Very truly yours,

Robert J. Barrett
Vice President, Operations
Indian Point 3 Nuclear Power Plant

References:

1. Indian Point 3 letter (IPN-00-042) to NRC, "Proposed Changes to the Administrative Section of Technical Specifications," dated June 7, 2000.

2. Indian Point 3 letter (IPN-00-062) to NRC, "Proposed Change to Section 6.14 of the Administrative Section of Technical Specifications," dated September 6, 2000.

3. Indian Point 3 letter (IPN-00-064) to NRC, "Proposed Technical Specification Amendment to Extend the Surveillance Frequency for the Fuel Storage Building Emergency Ventilation System," dated September 7, 2000.

4. Indian Point 3 letter (IPN-99-123) to NRC, "Response to NRC Generic Letter 99-02 and Application for Technical Specification Amendment for Laboratory Testing of Nuclear Grade Activated Charcoal," dated November 29, 1999.

5. Indian Point 3 letter (IPN-00-063) to NRC, "Proposed Change to the Technical Specifications Regarding Revised Main Feedwater Isolation Design," dated September 7, 2000.

6. Indian Point 3 letter (IPN-00-065) to NRC, "Proposed One Time Change to Technical Specifications Regarding the Replacement of 125VDC Station Batteries 31 and 32," dated September 7, 2000.

7. Indian Point 3 letter (IPN-01-012) to NRC, "Proposed One Time Change to the Technical Specification Regarding Allowed Outage Time Associated with One Diesel Generator or Any Diesel Fuel Oil System," dated February 14, 2001.

Attachment

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Docket 50-286
ATTACHMENT 1 TO IPN-01-030

Technical Specification Pages Associated With The Type A Integrated Leak Rate Test Interval Change

Insert	Delete
Section 5.0, Amendment (Pages 5.0-1 to 5.0-38)	Section 5.0, Amendment 205 (Pages 5.0-1 to 5.0-38)

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager 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 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. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator 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 unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

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, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR and Quality Assurance Plan, as appropriate;

b. The plant manager 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;

c. The corporate officer with direct responsibility for the plant 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

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.

(continued)

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

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 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 unit is in MODE 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 minimum requirements.

d. A radiation protection 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 unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technician, auxiliary operators, and key maintenance personnel).

   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:

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

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, shift turnover can be included in the break;

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 manager 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 periodically by the plant manager 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 assistant operations manager shall hold an SRO license.

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. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position must be manned in Mode 1, 2, 3 or 4 only.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the following:

a. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; and

b. The operations manager shall meet or exceed the minimum qualifications of ANSI N18.1-1971 except for the SRO license requirement which shall be in accordance with Technical Specification 5.2.2.f.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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 0, Appendix A, November 1972;

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;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. 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 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.

c. Licensee initiated changes to the ODCM:

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

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

(b) 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;

2. Shall become effective after the approval of the plant manager; and

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

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

in which any change in 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 (i.e., month and year) the change was implemented.

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 the following:

a. Residual Heat Removal System;

b. Cross Connect Between Low Head Recirculation System and High Head Safety Injection System;

c. High Head Safety Injection system (partial);

d. Reactor Coolant Sampling System;

e. Post Accident Containment Air Sampling System;

f. Volume Control Tank (including Reactor Coolant Pump seal return line);

g. Containment Hydrogen Monitoring system.

The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

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

(continued)

5.5 Programs and Manuals

5.5.3 Post Accident Sampling

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.

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 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 concentration values 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;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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;

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;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   b. For iodine-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to dose rate of 1500 mrems/yr to any organ.

h. 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;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, 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

j. 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 4.1.5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. The program shall include inspection frequencies and acceptance criteria. The inspection frequency will ensure that each reactor coolant pump flywheel is surface and volumetrically inspected within 10 years after a flywheel is placed in service following inspection.

(continued)

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 as follows:

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

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 ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the inservice inspection of SG tubes to assure the continued integrity of the Reactor Coolant System pressure boundary and shall include the following:

a. SG Selection and SG Tube Sample Size

The minimum sample size shall be in conformance with the requirements specified in Table 5.5-1. Selection and testing of steam generator tubes shall be made on the following basis:

1. At the first inservice inspection subsequent to the pre-service inspection, six percent of the tubes in each of two steam generators shall be inspected as a minimum.

2. At the second inservice inspection subsequent to the pre-service inspection, twelve percent of the tubes in one of the two steam generators not inspected during the first inservice inspection shall be inspected as a minimum.

3. At the third inservice inspection subsequent to the pre-service inspection, twelve percent of the tubes in the steam generator not inspected during the first two inservice inspections shall be inspected as a minimum.

4. Fourth and subsequent inservice inspections may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in like manner. Under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequences should be modified to inspect the steam generator with the most severe conditions.

5. Unscheduled inspections should be conducted on the affected steam generator(s) in accordance with the first sample

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

inspection specified in Table 5.5-1 in the event of primary-to-secondary tube leaks (not including leaks originated from tube-to-tube sheet welds) exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

b. SG Tube Selection Criteria

1. Tubes for the inspection should be selected on a random basis except where experience in similar plants with similar water chemistry indicates critical areas to be inspected.

2. The first sample inspection subsequent to the pre-service inspection should include all non-plugged tubes that previously had detectable wall penetration (> 20%) and should also include tubes in those areas where experience has indicated potential problems.

3. The second and third sample inspections in Table 5.5-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

4. In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetration to be included in the percentage calculation for the result categories in Table 5.5-1.

c. Inspection FREQUENCY

1. Inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

2. If the results of two consecutive inspections, not including the preservice inspection, all fall into the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

3. SR 3.0.2 is applicable to the Steam Generator Tube Surveillance Program test frequencies.

d. Classification of Test Results

1. Definitions:

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

Degradation means a service-induced cracking, wastage, wear or corrosion.

Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

% Degradation is an estimate % of the tube wall thickness affected or removed by degradation.

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

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

Tube Plugging Limit is the tube imperfection depth at or beyond which the tube must either be removed from service or repaired. This is considered to be an imperfection depth of 40%.

Sleeve Plugging Limit is the sleeve imperfection depth at or beyond which the sleeved tube must be removed from service or repaired. This is considered to be an imperfection depth of 40% for tube sleeves.

Tube Inspection is a full length inspection for the initial 3% sample specified in Table 5.5-1. Supplemental sample inspections (after the initial 3% sample) may be limited to a partial length inspection concentrating on those locations where degradation has been found.

2. Results Classifications

The results of each sampling examination of a steam generator shall be classified into the following three categories:

Category C-1: Less than 5% of the total tubes inspected are degraded tubes and none are defective.

Category C-2: One or more but not more than 1% of the total tubes inspected are defective or between 5 and 10% of the tubes inspected are degraded tubes.

Category C-3: More than 10% of the total tubes inspected are degraded or more than 1% of the tubes inspected are defective.

e. Corrective Action

1. The inspection result classification and the corresponding required action are specified in Table 5.5-1.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

2. All leaking tubes and defective tubes should be plugged or repaired.

3. Results of steam generator tube inspections which fall into Category C-3 of Table 5.5-1 require notification of the NRC within 15 days of this determination.

4. NRC approval prior to startup is required when SG Tube Inspections identify Category C-3 degradation or defects in more than one SG.

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-1	Acceptable for Service	C-1	Acceptable for Service	C-1	Acceptable for Service
C-2	Plug or Repair defective tubes AND Inspect additional 2S tubes in this SG	C-1	Acceptable for Service	N/A	N/A
		C-2	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG	C-1	Acceptable for Service
				C-2	Plug or Repair defective tubes AND Acceptable for Service
				C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG
		C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	N/A	N/A

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	All other SGs C-1	Acceptable for Service	N/A	
		Some SGs C-2 AND No other SG C-3	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG		
		Other SG C-3	Inspect all tubes in all SGs. AND Plug or Repair defective tubes AND Report and NRC Approval required prior to startup		

Sample Size shall consist of a minimum of S tubes per Steam Generator (SG)

$S=3(N/n)\%$

where:

N is the number of steam generators in the plant

n is the number of steam generators inspected during an examination

Result Classifications (C-1, C-2 and C-3) are defined in Section 5.5.8.d.

(continued)

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program 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 condenser hot wells 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 the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

This program provides controls for implementation of required testing the ventilation filter function for the Fuel Storage Building Emergency Ventilation System, Control Room Ventilation System, Containment Fan Cooler Units, and Containment Purge System.

Applicable tests described in Specifications 5.5.10.a, 5.5.10.b, 5.5.10.c and 5.5.10.d shall be performed:

1) After 720 hours of charcoal adsorber use since the last test; and,

2) Every 24 months for the Fuel Storage Building Emergency Ventilation System, Control Room Ventilation System, and Containment Fan Cooler Units; and,

3) Every 18 months for the Containment Purge System; and,

4) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; and,

5) After any structural maintenance on the system housing that could alter system integrity; and,

6) After significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

SR 3.0.2 is applicable to the Ventilation Filter Testing Program.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each system that an inplace test of the high efficiency particulate air (HEPA) filters shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Fuel Storage Building Emergency Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Containment Purge System	≥ 99%	90% to 110% of design operating rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each system that an inplace test of the charcoal adsorber shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Fuel Storage Building Emergency Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Containment Purge System	≥ 99%	90% to 110% of design operating rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each system that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide removal efficiency specified below when tested at the conditions specified below.

Ventilation System	Methyl iodide removal efficiency (%):	Methyl iodide inlet concentration (mg/m^3):	Flow velocity equivalent to following flow rate (cfm):	Temperature (degrees F):	Relative Humidity (%):
Fuel Storage Building Emergency Ventilation System	≥ 90	0.05 to 0.15	80% to 120% of design accident rate	≥ 125	≥ 95
Control Room Ventilation System	≥ 90	0.05 to 0.15	80% to 120% of design accident rate	≥ 125	≥ 95
Containment Fan Cooler Units	≥ 85	5 to 15	80% to 120% of design accident rate	≥ 250	≥ 95
Containment Purge System	≥ 90	*	80% to 120% of design operating rate	*	*

* Per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each system that the pressure drop across the combined HEPA filters, the demisters and prefilters (if installed), and the charcoal adsorbers is less than the value specified below when tested at the flowrate specified below.

Ventilation System	Delta P (inches wg)	Flowrate (cfm):
Fuel Storage Building Emergency Ventilation System	6	≥ 90% of design accident rate
Control Room Ventilation System	6	≥ 90% of design accident rate
Containment Fan Cooler Units	6	≥ 90% of design accident rate

(continued)

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The quantities of radioactivity in gas and liquid radwaste storage tanks shall be determined in accordance with methodology and parameters specified in the ODCM.

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 shall be limited to less than or equal to 50,000 curies noble gases (considered as DOSE EQUIVALENT Xe-133); 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 or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

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.

(continued)

5.5 Programs and Manuals

5.5.12 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 for the DG fuel oil onsite storage tanks and the DG reserve fuel oil storage tanks. 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. Verification of the acceptability of new fuel oil for use prior to addition to the DG fuel oil onsite storage tanks by determining that the fuel oil has:

   1. Relative density within the limits of 0.83 to 0.89,

   2. kinematic viscosity within the limits of 1.8 to 5.8, and

   3. a clear and bright appearance with proper color

b1. Verification of the acceptability of the fuel oil in the onsite storage tanks and the reserve storage tanks every 92 days by verifying that the properties of the fuel oil in the tanks, other than those addressed in item a., are within limits for ASTM2D fuel oil. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.

or

b2. Verification of the acceptability of each new fuel addition made subsequent to the last verification made in accordance with item b1. by verifying within 31 days following the addition that the properties of the new fuel oil, other than those properties addressed in item a. are within limits for ASTM 2D fuel oil.

c. Verification every 92 days that total particulate concentration of the fuel oil in the onsite and reserve storage tanks is less than or equal to 10 mg/l when tested in accordance with ASTM D-2276, Method A-2 or A-3. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.

(continued)

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

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

5.5.13 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 requires NRC approval pursuant to 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 do not meet the criteria of Specification 5.5.13.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).

5.5.14 Safety Function 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 actions may 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:

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

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 the system(s) supported by the inoperable support system is also inoperable; or

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

c. A required system redundant to the 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.

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage 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. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995" as modified by the following exception:

> ANS 56.8-1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested.
>
> NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the December 2, 1990 Type A test shall be performed no later than December 1, 2005.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq 1.1\ P_a$.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

cooler unit when pressurized at $\geq 1.1\ P_a$. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The peak calculated containment internal pressure for the design basis main steam line break, Pa, is 42.40 psig. The minimum test pressure is 42.42 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.1% of primary containment air weight per day.

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

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent ≥ 100 mrems and the associated collective 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, 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, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from 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 Annual Radiological Environmental Operating 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.

The Annual Radiological Environmental Operating Report covering the operation of the unit 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 (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

5.6.3 Radioactive Effluent Release Report

--------------------------------NOTE--------------------------------
A single submittal may be made for a multiple unit station. The submittal shall 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.

The Radioactive Effluent Release Report covering the operation of the unit 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 ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer 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.

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:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Specification 3.1.1, SHUTDOWN MARGIN;

2. Specification 3.1.3, Moderator Temperature Coefficient;

3. Specification 3.1.5, Shutdown Bank Insertion Limits;

4. Specification 3.1.6, Control Bank Insertion Limits;

5. Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);

6. Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);

7. Specification 3.2.3, AXIAL FLUX DIFFERENCE (AFD); and

8. Specification 3.9.1, Boron Concentration.

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 following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Specifications 3. 1.5, Shutdown Bank Insertion Limits, 3.1.6, Control Bank Insertion Limits, and 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor);

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES, TOPICAL REPORT," September 1974 (W Proprietary). (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control);

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

3a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3b. WCAP-9561-P-A ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS, SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3c. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).(Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z))); and

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor).

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 midcycle revisions or supplements, shall be provided for each reload cycle to the NRC.

5.6.6 NOT USED

(continued)

5.6 Reporting Requirements

5.6.7 Post Accident Monitoring Instrumentation (PAM) Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the next 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.

5.6.8 Steam Generator Tube Inspection Report

The number of tubes plugged or repaired in each steam generator during each inservice inspection of steam generator tubes shall be reported to the Commission within 15 days following the inspection.

Complete results of the steam generator tube inservice inspections shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 5.5.8. This report shall include:

a. Number and extent of tubes inspected.

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

c. Identification of the tubes plugged and the tubes repaired.

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 > 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 protection 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 protection manager in the RWP.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels $\geq$ 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.

5.7.3 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.

Docket 50-286
ATTACHMENT 2 TO IPN-01-030

Technical Specification Pages Associated With Fuel Storage Emergency Ventilation System Charcoal Testing Interval

Insert	Delete
Section 5.0, Amendment (Pages 5.0-1 to 5.0-38)	Section 5.0, Amendment 205 (Pages 5.0-1 to 5.0-38)

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager 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 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. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator 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 unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

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, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR and Quality Assurance Plan, as appropriate;

b. The plant manager 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;

c. The corporate officer with direct responsibility for the plant 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

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.

(continued)

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

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 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 unit is in MODE 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 minimum requirements.

d. A radiation protection 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 unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technician, auxiliary operators, and key maintenance personnel).

   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:

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

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, shift turnover can be included in the break;

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 manager 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 periodically by the plant manager 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 assistant operations manager shall hold an SRO license.

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. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position must be manned in Mode 1, 2, 3 or 4 only.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the following:

a. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; and

b. The operations manager shall meet or exceed the minimum qualifications of ANSI N18.1-1971 except for the SRO license requirement which shall be in accordance with Technical Specification 5.2.2.f.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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 0, Appendix A, November 1972;

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;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. 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 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.

c. Licensee initiated changes to the ODCM:

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

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

   (b) 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;

2. Shall become effective after the approval of the plant manager; and

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

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

in which any change in 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 (i.e., month and year) the change was implemented.

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 the following:

a. Residual Heat Removal System;

b. Cross Connect Between Low Head Recirculation System and High Head Safety Injection System;

c. High Head Safety Injection system (partial);

d. Reactor Coolant Sampling System;

e. Post Accident Containment Air Sampling System;

f. Volume Control Tank (including Reactor Coolant Pump seal return line);

g. Containment Hydrogen Monitoring system.

The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

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

(continued)

5.5 Programs and Manuals

5.5.3 Post Accident Sampling

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.

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 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 concentration values 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;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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;

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;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   b. For iodine-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to dose rate of 1500 mrems/yr to any organ.

h. 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;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, 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

j. 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 4.1.5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. The program shall include inspection frequencies and acceptance criteria. The inspection frequency will ensure that each reactor coolant pump flywheel is surface and volumetrically inspected within 10 years after a flywheel is placed in service following inspection.

(continued)

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 as follows:

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

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 ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the inservice inspection of SG tubes to assure the continued integrity of the Reactor Coolant System pressure boundary and shall include the following:

a. SG Selection and SG Tube Sample Size

The minimum sample size shall be in conformance with the requirements specified in Table 5.5-1. Selection and testing of steam generator tubes shall be made on the following basis:

1. At the first inservice inspection subsequent to the preservice inspection, six percent of the tubes in each of two steam generators shall be inspected as a minimum.

2. At the second inservice inspection subsequent to the preservice inspection, twelve percent of the tubes in one of the two steam generators not inspected during the first inservice inspection shall be inspected as a minimum.

3. At the third inservice inspection subsequent to the preservice inspection, twelve percent of the tubes in the steam generator not inspected during the first two inservice inspections shall be inspected as a minimum.

4. Fourth and subsequent inservice inspections may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in like manner. Under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequences should be modified to inspect the steam generator with the most severe conditions.

5. Unscheduled inspections should be conducted on the affected steam generator(s) in accordance with the first sample

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

inspection specified in Table 5.5-1 in the event of primary-to-secondary tube leaks (not including leaks originated from tube-to-tube sheet welds) exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

b. SG Tube Selection Criteria

1. Tubes for the inspection should be selected on a random basis except where experience in similar plants with similar water chemistry indicates critical areas to be inspected.

2. The first sample inspection subsequent to the pre-service inspection should include all non-plugged tubes that previously had detectable wall penetration (> 20%) and should also include tubes in those areas where experience has indicated potential problems.

3. The second and third sample inspections in Table 5.5-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

4. In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetration to be included in the percentage calculation for the result categories in Table 5.5-1.

c. Inspection FREQUENCY

1. Inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

2. If the results of two consecutive inspections, not including the preservice inspection, all fall into the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

3. SR 3.0.2 is applicable to the Steam Generator Tube Surveillance Program test frequencies.

d. Classification of Test Results

1. Definitions:

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

Degradation means a service-induced cracking, wastage, wear or corrosion.

Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

% Degradation is an estimate % of the tube wall thickness affected or removed by degradation.

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

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

Tube Plugging Limit is the tube imperfection depth at or beyond which the tube must either be removed from service or repaired. This is considered to be an imperfection depth of 40%.

Sleeve Plugging Limit is the sleeve imperfection depth at or beyond which the sleeved tube must be removed from service or repaired. This is considered to be an imperfection depth of 40% for tube sleeves.

Tube Inspection is a full length inspection for the initial 3% sample specified in Table 5.5-1. Supplemental sample inspections (after the initial 3% sample) may be limited to a partial length inspection concentrating on those locations where degradation has been found.

2. Results Classifications

The results of each sampling examination of a steam generator shall be classified into the following three categories:

Category C-1: Less than 5% of the total tubes inspected are degraded tubes and none are defective.

Category C-2: One or more but not more than 1% of the total tubes inspected are defective or between 5 and 10% of the tubes inspected are degraded tubes.

Category C-3: More than 10% of the total tubes inspected are degraded or more than 1% of the tubes inspected are defective.

e. Corrective Action

1. The inspection result classification and the corresponding required action are specified in Table 5.5-1.

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

2. All leaking tubes and defective tubes should be plugged or repaired.

3. Results of steam generator tube inspections which fall into Category C-3 of Table 5.5-1 require notification of the NRC within 15 days of this determination.

4. NRC approval prior to startup is required when SG Tube Inspections identify Category C-3 degradation or defects in more than one SG.

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-1	Acceptable for Service	C-1	Acceptable for Service	C-1	Acceptable for Service
C-2	Plug or Repair defective tubes AND Inspect additional 2S tubes in this SG	C-1	Acceptable for Service	N/A	N/A
		C-2	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG	C-1	Acceptable for Service
				C-2	Plug or Repair defective tubes AND Acceptable for Service
				C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG
		C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	N/A	N/A

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	All other SGs C-1	Acceptable for Service	N/A	
		Some SGs C-2 AND No other SG C-3	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG		
		Other SG C-3	Inspect all tubes in all SGs. AND Plug or Repair defective tubes AND Report and NRC Approval required prior to startup		

Sample Size shall consist of a minimum of S tubes per Steam Generator (SG)

$S=3(N/n)\%$

where:

N is the number of steam generators in the plant

n is the number of steam generators inspected during an examination

Result Classifications (C-1, C-2 and C-3) are defined in Section 5.5.8.d.

(continued)

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program 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 condenser hot wells 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 the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

This program provides controls for implementation of required testing the ventilation filter function for the Fuel Storage Building Emergency Ventilation System, Control Room Ventilation System, Containment Fan Cooler Units, and Containment Purge System.

Applicable tests described in Specifications 5.5.10.a, 5.5.10.b, 5.5.10.c and 5.5.10.d shall be performed:

1) After 720 hours of charcoal adsorber use since the last test for the above systems except the Fuel Storage Building Emergency Ventilation System; and,

2) After 1440 hours of charcoal adsorber use since the last test for the Fuel Storage Building Emergency Ventilation System; and,

3) Every 24 months for the Fuel Storage Building Emergency Ventilation System, Control Room Ventilation System, and Containment Fan Cooler Units; and,

4) Every 18 months for the Containment Purge System; and,

5) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; and,

6) After any structural maintenance on the system housing that could alter system integrity; and,

7) After significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

SR 3.0.2 is applicable to the Ventilation Filter Testing Program.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each system that an inplace test of the high efficiency particulate air (HEPA) filters shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Fuel Storage Building Emergency Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Containment Purge System	≥ 99%	90% to 110% of design operating rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each system that an inplace test of the charcoal adsorber shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Fuel Storage Building Emergency Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Containment Purge System	≥ 99%	90% to 110% of design operating rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each system that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide removal efficiency specified below when tested at the conditions specified below.

Ventilation System	Methyl iodide removal efficiency (%):	Methyl iodide inlet concentration (mg/m^3):	Flow velocity equivalent to following flow rate (cfm):	Temperature (degrees F):	Relative Humidity (%):
Fuel Storage Building Emergency Ventilation System	≥ 90	0.05 to 0.15	80% to 120% of design accident rate	≥ 125	≥ 95
Control Room Ventilation System	≥ 90	0.05 to 0.15	80% to 120% of design accident rate	≥ 125	≥ 95
Containment Fan Cooler Units	≥ 85	5 to 15	80% to 120% of design accident rate	≥ 250	≥ 95
Containment Purge System	≥ 90	*	80% to 120% of design operating rate	*	*

* Per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each system that the pressure drop across the combined HEPA filters, the demisters and prefilters (if installed), and the charcoal adsorbers is less than the value specified below when tested at the flowrate specified below.

Ventilation System	Delta P (inches wg)	Flowrate (cfm):
Fuel Storage Building Emergency Ventilation System	6	≥ 90% of design accident rate
Control Room Ventilation System	6	≥ 90% of design accident rate
Containment Fan Cooler Units	6	≥ 90% of design accident rate

(continued)

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The quantities of radioactivity in gas and liquid radwaste storage tanks shall be determined in accordance with methodology and parameters specified in the ODCM.

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 shall be limited to less than or equal to 50,000 curies noble gases (considered as DOSE EQUIVALENT Xe-133); 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 or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

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.

(continued)

5.5 Programs and Manuals

5.5.12 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 for the DG fuel oil onsite storage tanks and the DG reserve fuel oil storage tanks. 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. Verification of the acceptability of new fuel oil for use prior to addition to the DG fuel oil onsite storage tanks by determining that the fuel oil has:

   1. Relative density within the limits of 0.83 to 0.89,

   2. kinematic viscosity within the limits of 1.8 to 5.8, and

   3. a clear and bright appearance with proper color

b1. Verification of the acceptability of the fuel oil in the onsite storage tanks and the reserve storage tanks every 92 days by verifying that the properties of the fuel oil in the tanks, other than those addressed in item a., are within limits for ASTM2D fuel oil. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.

   or

b2. Verification of the acceptability of each new fuel addition made subsequent to the last verification made in accordance with item b1. by verifying within 31 days following the addition that the properties of the new fuel oil, other than those properties addressed in item a. are within limits for ASTM 2D fuel oil.

c. Verification every 92 days that total particulate concentration of the fuel oil in the onsite and reserve storage tanks is less than or equal to 10 mg/l when tested in accordance with ASTM D-2276, Method A-2 or A-3. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.

(continued)

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

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

5.5.13 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 requires NRC approval pursuant to 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 do not meet the criteria of Specification 5.5.13.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).

5.5.14 Safety Function 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 actions may 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:

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

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 the system(s) supported by the inoperable support system is also inoperable; or

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

c. A required system redundant to the 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.

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage 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. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995" as modified by the following exception:

> ANS 56.8-1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq 1.1\ P_a$.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan cooler unit when pressurized at $\geq 1.1\ P_a$. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The peak calculated containment internal pressure for the design basis main steam line break, Pa, is 42.40 psig. The minimum test pressure is 42.42 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.1% of primary containment air weight per day.

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

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent ≥ 100 mrems and the associated collective 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, 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, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from 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 Annual Radiological Environmental Operating 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.

The Annual Radiological Environmental Operating Report covering the operation of the unit 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 (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal shall 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.

The Radioactive Effluent Release Report covering the operation of the unit 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 ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer 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.

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:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Specification 3.1.1, SHUTDOWN MARGIN;

2. Specification 3.1.3, Moderator Temperature Coefficient;

3. Specification 3.1.5, Shutdown Bank Insertion Limits;

4. Specification 3.1.6, Control Bank Insertion Limits;

5. Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);

6. Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);

7. Specification 3.2.3, AXIAL FLUX DIFFERENCE (AFD); and

8. Specification 3.9.1, Boron Concentration.

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 following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Specifications 3. 1.5, Shutdown Bank Insertion Limits, 3.1.6, Control Bank Insertion Limits, and 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor);

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES, TOPICAL REPORT," September 1974 (W Proprietary). (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control);

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

3a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3b. WCAP-9561-P-A ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS, SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3c. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).(Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z))); and

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor).

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 midcycle revisions or supplements, shall be provided for each reload cycle to the NRC.

5.6.6 NOT USED

(continued)

5.6 Reporting Requirements

5.6.7 Post Accident Monitoring Instrumentation (PAM) Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the next 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.

5.6.8 Steam Generator Tube Inspection Report

The number of tubes plugged or repaired in each steam generator during each inservice inspection of steam generator tubes shall be reported to the Commission within 15 days following the inspection.

Complete results of the steam generator tube inservice inspections shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 5.5.8. This report shall include:

a. Number and extent of tubes inspected.

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

c. Identification of the tubes plugged and the tubes repaired.

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 > 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 protection 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 protection manager in the RWP.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, 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.

5.7.3 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.

Docket 50-286
ATTACHMENT 3 TO IPN-01-030

Technical Specification Pages Associated With The Main Feedwater Isolation Modification For Design Basis Events

Insert	Delete
Table of Contents, Amendment (Pages i to v)	Table of Contents, Amendment 205 (Pages i to v)
Section 3.7.3, Amendment (Pages 3.7.3-1 to 3.7.3-3)	Section 3.7.3, Amendment 205 (Pages 3.7.3-1 to 3.7.3-3)

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3.7 PLANT SYSTEMS

3.7.3 Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs), Main Feedwater Inlet Isolation Valves (MFIIVs) and Main Feedwater (MF) Low Flow Bypass Valves

LCO 3.7.3 Two MBFPDVs, four MFRVs, four MFIIVs and eight MF low flow bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when each main feedwater and bypass line is isolated by a closed and de-activated motor/air operated valve or isolated by a closed manual valve.

ACTIONS

-------------------------------------NOTE-------------------------------------

Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both MBFPDVs inoperable.	A.1 Close or isolate MBFPDV.	72 hours
	<u>AND</u>	
	A.2 Verify MBFPDV is closed or isolated.	Once per 7 days
B. One or more MFRVs inoperable.	B.1 Close or isolate MFRV.	72 hours
	<u>AND</u>	
	B.2 Verify MFRV is closed or isolated.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more MFIIVs inoperable.	C.1 Close or isolate MFIIV. AND C.2 Verify MFIIV is closed or isolated.	72 hours Once per 7 days
D. One or more MF low flow bypass valves inoperable.	D.1 Close or isolate bypass valve. AND D.2 Verify bypass valve is closed or isolated.	72 hours Once per 7 days
E. Two valves in series in the same flow path inoperable.	E.1 Isolate affected flow path.	8 hours
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.3.1	Verify each MBFPDV, MFRV, MFIIV and MF low flow bypass valve closes on an actual or simulated actuation signal within the following limits: a. MBFPDV closure time ≤ 122 seconds; b. MFRV closure time ≤ 10 seconds; and, c. MFIIV closure time ≤ 120 seconds d. MFRV Low Flow Bypass valve closure time 1. primary ≤ 10 seconds 2. backup ≤ 120 seconds.	In accordance with the Inservice Testing Program

Docket 50-286
ATTACHMENT 4 TO IPN-01-030

Technical Specification Pages Associated With
One Time Change For Replacing Station Batteries

Insert	Delete
Section 3.8.4 Amendment (Pages 3.8.4-1 to 3.8.4-5)	Section 3.8.4 Amendment 205 (Pages 3.8.4-1 to 3.8.4-4)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

LCO 3.8.4 The following four DC electrical power subsystems shall be OPERABLE:

Battery 31 and associated Battery Charger;
Battery 32 and associated Battery Charger;
Battery 33 and associated Battery Charger; and
Battery 34.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DC electrical power subsystem 34 inoperable.	A.1 Declare Inverter 34 inoperable and take Required Actions specified in LCO 3.8.7, Inverters-Operating.	2 hours
B. One DC electrical power subsystem (31 or 32 or 33) inoperable.	B.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours(1)
C. Required Action and Associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

(1) On a one-time (per battery) only basis for station batteries 31 and 32, the batteries may be inoperable for up to 10 days each, as necessary, to allow on-line replacement of the batteries. The time-period during which this allowance may be exercised will end on May 31, 2002. The following addtional requirement shall also be met to invoke this extended one-time allowed outage time: No risk significant planned maintenance or testing activities, which may impact AC or DC normal or emergency distribution sources of ESF systems, shall be performed during this replacement period.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage on float charge is within the following limits: a. ≥ 123.5 V for batteries 31 and 32; and b. ≥ 127.8 V for batteries 33 and 34.	31 days
SR 3.8.4.2	--------------------NOTE--------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify each battery charger supplies its associated battery at the voltage and current adequate to demonstrate battery charger capability requirements are met.	24 months
SR 3.8.4.3	--------------------NOTES-------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.[2] -- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (CONTINUED)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	-------------------NOTE--------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating
SR 3.8.4.5	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	24 months

SURVEILLANCE REQUIREMENTS (CONTINUED)

(2) This battery surveillance may be performed on a one-time only basis during replacement of station batteries 31 and 32 when the unit is in Mode 1, 2, 3 or 4 in order to support the one-time allowed outage time change of 10 days, as indicated in section 3.4.B. This testing shall be done when the battery is disconnected from the DC bus.

Docket 50-286
ATTACHMENT 5 TO IPN-01-030

Technical Specification Pages Associated With One Time Change For Diesel Generator or Diesel Fuel Oil System Allowed Outage Time

Insert	Delete
Section 3.8.1 and 3.8.3, Amendment (Pages 3.8.1-1 to 3.8.1-12 and Pages 3.8.3-1 to 3.8.3-6)	Section 3.8.1 and 3.8.3, Amendment 205 (Pages 3.8.1-1 to 3.8.1-10 and Pages 3.8.3-1 to 3.8.3-5)

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Electrical Power Distribution System; and

b. Three diesel generators (DGs) (31, 32 and 33) capable of supplying the onsite power distribution subsystem(s)

------------------------------ NOTE ------------------------------
The 138 kV circuit is considered inoperable whenever the automatic transfer function for the 6.9 kV buses is disabled.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour
		AND
	AND	Once per 8 hours thereafter
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)		--------NOTE--------- Only required if 13.8 kV offsite circuit is supplying 6.9 kV bus 5 or 6 and the Unit Auxiliary Transformer is supplying 6.9 kV bus 1, 2, 3 or 4. -----------------------	
	A.2	Verify automatic transfer of 6.9 kV buses 1, 2, 3, and 4 to 6.9 kV bus 5 and 6 is disabled.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	A.3	Declare inoperable required feature(s) with no offsite power automatically available when its redundant required feature(s) is inoperable.	24 hours from discovery of no automatically available offsite power to one train concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	A.4	Restore offsite circuit to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for the offsite circuits.	1 hour AND Once per 8 hours thereafter
	AND	
	B.2 Declare inoperable the required features supported by the inoperable DG when its required redundant feature is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature
	AND	
	B.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure. OR	24 hours
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DGs.	24 hours
	AND	
	B.4 Restore DG to OPERABLE status.	72 hours (1)

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required features inoperable when its redundant required feature is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours
D. One offsite circuit inoperable. AND One DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating," when Condition D is entered with no offsite or DG AC power source automatically available to any train. ------------------------	
	D.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	D.2 Restore DG to OPERABLE status.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more DGs inoperable.	E.1 Restore at least two DGs to OPERABLE status.	2 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours
G. One or more offsite circuits and two DGs inoperable.	G.1 Enter LCO 3.0.3.	Immediately
H. Two offsite circuits and one or more DGs inoperable.	H.1 Enter LCO 3.0.3.	Immediately

(1) Each of 31, 32 or 33 emergency diesel generator (EDG) fuel oil storage tanks (FOSTs) may be inoperable and its associated EDG may be declared technically inoperable, but available and capable of automatic start, for up to 14 days, one time only if needed, during 2001 prior to August 31, 2001. This condition may only be invoked to inspect/repair each of 31, 32 or 33 EDG FOSTs once, if deemed necessary based on concerns with water inleakage. The following additional requirements shall also be met for each FOST inspection/repair to invoke its extended one-time allowed outage time:
   (1) performance of off-site power source switching or maintenance evolutions for technical specification required offsite power sources shall not be scheduled during these 31, 32 or 33 FOST outages, and
   (2) these 31, 32 or 33 FOST outages shall not be scheduled during predicted severe weather.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	--------------------NOTE--------------------- All DG starts may be preceded by an engine prelube period. --- Verify each DG starts from standby conditions and achieves: a. in ≤ 10 seconds, voltage ≥ 422 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 422 V and ≤ 500V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-------------------NOTES------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This SR shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 1575 kW and ≤ 1750 kW.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 115 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from DG storage tank to the day tank.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1 or 2. -- Verify manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit.	24 months
SR 3.8.1.8	------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1 or 2. 2. Only required to be met if 138 kV offsite circuit is supplying 6.9 kV bus 5 and 6 and the Unit Auxiliary Transformer is supplying 6.9 kV bus 2 or 3. -- Verify automatic transfer of AC power for 6.9 kV buses 2 and 3 from the unit auxiliary transformer to 6.9 kV buses 5 and 6.	24 months

(continued)

SR 3.8.1.9	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1 or 2. -- Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal except: a. Engine overspeed; b. Low lube oil pressure; and c. Overcrank relay.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-------------------NOTES-------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. -- Verify each DG operating at a power factor ≤ 0.9 operates for ≥ 8 hours: a. For ≥ 105 minutes loaded ≥ 1837 kW and ≤ 1925 kW; and b. For the remaining hours of the test loaded ≥ 1575 kW and ≤ 1750 kW.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	----------NOTE---------- Load timers associated with equipment that has automatic initiation capability disabled are not required to be operable. ------------------------------ Verify each time delay relay functions within the required design interval.	18 months
SR 3.8.1.12	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. This SR may be performed on safeguards power trains one at a time, or simultaneously. Appropriate plant conditions must be established when testing three safeguards power trains simultaneously. ------------------------------	(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.12 (continued) Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in $\leq$ 10 seconds, 2. energizes auto-connected emergency loads through individual load timers, 3. achieves steady state voltage $\geq$ 422 V and $\leq$ 500 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-------------------NOTE------------------- 1. All DG starts may be preceded by an engine prelube period. 2. Performance of SR 3.8.1.12 may be used to satisfy the requirements of this SR if all three diesel generators are started simultaneously. --	
	Verify when started simultaneously from standby condition, each DG achieves: a. in ≤ 10 seconds, voltage ≥ 422 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 422 V and ≤ 500V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	10 years

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil and Starting Air

LCO 3.8.3 The stored diesel fuel oil and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-------------------------------------NOTE-------------------------------------

Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Only applicable in MODES 1, 2, 3 and 4. ---------------------- One or more DGs with usable fuel oil in associated DG fuel oil storage tank < 5365 gal.[1]	A.1	Declare associated DG inoperable.	Immediately

(continued)

(1) If this condition is met due to a deliberate one-time inspection/repair due to water in-leakage, of 31, 32 or 33 EDG FOST, then the note (1) of LCO 3.8.1.B.4 completion time applies.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- Only applicable in MODES 5 and 6 and during movement of irradiated fuel. ----------------------- Total combined usable fuel oil in DG fuel oil storage tanks associated with the operable DG(s) < 5365 gal.	B.1 Declare all DGs inoperable.	Immediately
C. ---------NOTE----------- Only applicable in MODES 1, 2, 3 and 4. ----------------------- Total useable fuel oil in reserve storage tank(s) < 26,826 gal.	C.1 Declare all DGs inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more DG fuel oil storage tanks or reserve fuel oil storage tanks with fuel oil total particulates not within limits.	D.1	Restore fuel oil total particulates within limit.	7 days for DG fuel oil storage tank AND 30 days for reserve fuel oil storage tank
E. One or more DG fuel oil storage tanks or reserve fuel oil storage tanks with fuel oil properties other than particulates not within limits.	E.1	Restore fuel oil properties to within limits.	30 days for DG fuel oil storage tank AND 60 days for reserve fuel oil storage tank
F. One or more DGs with starting air receiver pressure < 250 psig and ≥ 90 psig.	F.1	Restore starting air receiver pressure to ≥ 250 psig.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs diesel fuel oil or starting air subsystem not within limits for reasons other than Condition A, B, C, D, E, or F.	G.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.3.1	-------------------NOTE------------------- Only required in MODES 1, 2, 3 and 4. -- Verify reserve storage tank(s) contain ≥ 26,826 gal of fuel oil reserved for IP3 usage only.	24 hours
SR 3.8.3.2	Verify DG fuel oil storage tanks contain: a. Usable fuel oil volume ≥ 5365 gal in each storage tank when in MODES 1, 2, 3 and 4; and b. Total combined usable fuel oil volume ≥ 5365 gal in any DG fuel oil storage tank(s) that are associated with the operable DG(s) when in MODES 5 and 6 and during movement of irradiated fuel assemblies.	31 days
SR 3.8.3.3	Verify that fuel oil properties of new and stored fuel oil in the DG fuel oil storage tanks are tested and maintained in accordance with the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.4	--------------------NOTE-------------------- Only required in MODES 1, 2, 3 and 4. -- Verify that fuel oil properties in the reserve storage tank(s) are within limits specified in the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.5	Verify each DG air start receiver pressure is $\geq$ 250 psig.	31 days
SR 3.8.3.6	Check for and remove accumulated water from each DG fuel oil storage tank.	92 days