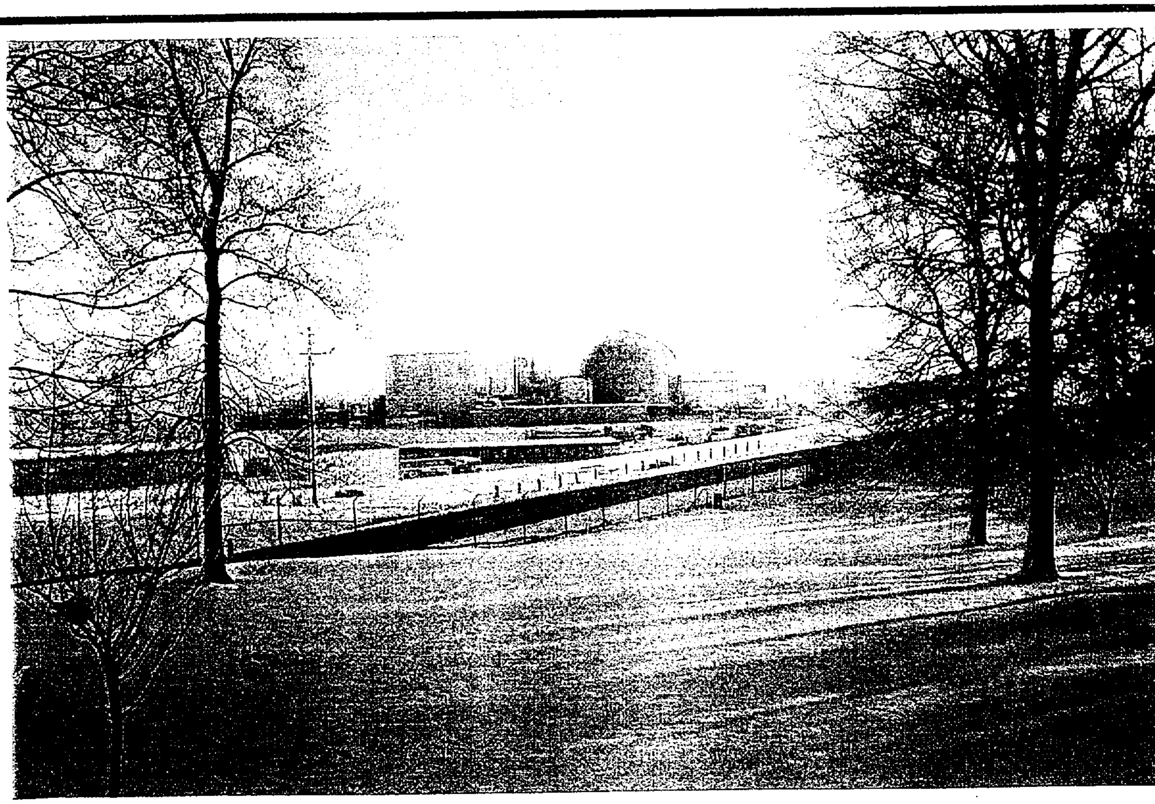
NORTH ANNA POWER STATION

Chapter 5.0
Administrative Controls



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NORTH ANNA POWER STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

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IMPROVED TECHNICAL SPECIFICATIONS

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 UFSAR/QA Plan;

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. A specified corporate officer 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.

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A total of four non-licensed operators shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A non-licensed operator, who may be one of the four assigned to a control room, shall be assigned to each reactor containing fuel.

b. 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.f 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.

c. 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.

d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel).

   The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

   Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

5.2 Organization

5.2.2 Unit Staff (continued)

e. The Superintendent Operations shall hold (or have previously held) a Senior Reactor Operator License for North Anna or a similar design Pressurized Water Reactor plant. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station.

f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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 3.1 (12/79 Draft) for comparable positions. Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase." The Superintendent-Radiological Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The SS, Assistant SS, Control Room Operator-Nuclear, and the individual providing advisory technical support to the unit operations shift crew, shall meet or exceed the minimum qualifications of 10 CFR 55.59(c) and 55.31(a)(4).

5.3.2 For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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 2, Appendix A, February 1978;

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

c. 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 Annual Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

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

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

2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

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

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of 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 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

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

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;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.10001-20.2402;

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

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

e. Determination of cumulative dose contributions from radioactive effluents for the current calender quarter and current calender year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology 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 from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/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;

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, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 5.2, 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 once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program surveillance frequency.

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. 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.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test Frequencies.

This program provides the controls for the inservice inspection of steam generator tubes to ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of steam generators is based on a modification of Regulatory Guide 1.83, Revision 1. This program shall include:

5.5.8.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.8-1.

5.5.8.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations > 20%, and

   2. Tubes in those areas where experience has indicated potential problems.

5.5 Programs and Manuals

5.5.8.2 Steam Generator Tube Sample Selection and Inspection

b. (continued)

3. A tube inspection (pursuant to Specification 5.5.8.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 5.5.8.2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

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

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

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

5.5 Programs and Manuals

5.5.8.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.8-2 at 40 month intervals fall into category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A major steam line or feedwater line break.

5.5 Programs and Manuals

5.5.8.4 Acceptance Criteria

a. As used in this Specification:

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

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

3. Degraded Tube means a tube containing imperfections > 20% of the nominal wall thickness caused by degradation.

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.8.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely around the U-bend to the top support.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy-current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspection.

5.5 Programs and Manuals

5.5.8.4 Acceptance Criteria (continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.8-2.

Table 5.5.8-1
Minimum Number of Steam Generators to Be Inspected
During Inservice Inspection

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspection	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the unit) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that 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 sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Table 5.5.8-2
Steam Generator Tube Inspection

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in SG	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in SG	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug defective tubes	N/A	N/A

S = 3[N/n}%   Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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 discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

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

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

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in general conformance with the frequencies and requirements of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
MCR/ESGR EVS	1000 ± 10% cfm
ECCS PREACS	Nominal accident flow for a single train actuation

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

a. (continued)

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
MCR/ESGR EVS	1000 ± 10% cfm
ECCS PREACS	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity specified below.

ESF Ventilation System	Penetration	RH
MCR/ESGR EVS	2.5%	70%
ECCS PREACS	5%	70%

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for the ECCS PREACS that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than 5" water gauge when tested in accordance with ANSI N510-1975 at a system flowrate ≤ 39,200 cfm.

e. Demonstrate for the MCR/ESGR EVS that the pressure drop across the combined HEPA filters, the demister filter, and the charcoal adsorbers is less than 4" water gauge when tested in accordance with ANSI N510-1975 at a system flowrate of 1000 cfm ± 10%.

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

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 Decay Tanks, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Decay Tanks 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 is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

5.5 Programs and Manuals

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

c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following outdoor 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 liquid radwaste ion exchanger system is less than the amount that would result in concentrations greater than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, excluding tritium, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents:

1. Refueling Water Storage Tank;
2. Casing Cooling Storage Tank;
3. PG Water Storage Tank;
4. Boron Recovery Test Tank; and
5. Any Outside Temporary Tank.

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.

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. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

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

1. an API gravity or an absolute specific gravity within limits,
2. kinematic viscosity within limits for ASTM 2D fuel oil, and
3. water and sediment $\leq$ 0.05%.

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

b. Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the stored fuel oil is $\leq$ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3; and

d. 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 require either of the following:

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

2. a change to the UFSAR 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13b 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 Programs and Manuals

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:

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, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), 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. When a

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish 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.

b. The Peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is 45 psig.

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

d. Leakage Rate acceptance criteria are:

   1. Prior to entering a MODE where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

      $\leq 0.60\ L_a$ for the Type B and Type C tests on a Maximum Path Basis and $\leq 0.75\ L_a$ for Type A tests.

      During operation where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

      $\leq 1.0\ L_a$ for overall containment leakage rate and $\leq 0.60\ L_a$ for the Type B and Type C tests on a Minimum Path Basis.

   2. Overall air lock leakage rate acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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

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

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 (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, 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 1 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)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements commensurate with the format in the ODCM. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Annual 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 Annual 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 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience 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:

   1. Safety Limits,

   2. Shutdown Margin,

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR)

a. (continued)

3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. Axial Flux Difference limits,
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
10. RCS Pressure, Temperature, and Flow DNB Limits, and
11. 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. VEP-FRD-42, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."
6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR)

b. (continued)

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY-REFERENCE CORE REPORT."

8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."

10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

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 upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report

b. (continued)

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

3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

d. (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the radiation protection shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

d. (continued)

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

f. (continued)
area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

IMPROVED TECHNICAL SPECIFICATIONS BASES

Chapter 5.0 does not have Bases

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

MARKUP AND JUSTIFICATION FOR DEVIATIONS

CTS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

manager

TSTF-65

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

manager

TSTF-65

The [Plant Superintendent] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

6.1.2 5.1.2 Table 6.2-1 (continued) 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. (1) (1)

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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

6.2.1 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.

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the [FSAR]; TSTF-65 ①

including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications

/QA Plan TSTF-65 TSTF-65

manager

6.2.1.b b. The [Plant Superintendent] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; TSTF-65

officer

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

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

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

, who may be one of the four assigned to the control room,

Table 6.2-1 a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator ⑥

A total of four

INSERT on ISTS page 5.0-3

(continued)

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5.2 Organization

5.2.2 Unit Staff (continued) INSERT FROM ISTS page 5.0-2 (6)

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

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.] (1)

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.~~ TSTF-258

Table 6.2-1 Notes c. b. 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 f 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. f TSTF-258 TSTF-65

6.2.2.c d. c. A ~~[Health Physics]~~ 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. TSTF-258

Table 6.2-1 (continued) NEW d. e. [Administrative procedures shall be ~~developed~~ personnel and implemented to limit the working hours of ~~unit staff~~ who perform safety related functions (e.g., licensed ~~SROs~~ Senior Reactor Operators (SROs), licensed ~~ROs~~ Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel). TSTF-258

~~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.~~ TSTF-258

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

(continued)

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5.2 Organization

5.2.2 Unit Staff (continued)

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;~~ TSTF 258

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

4. ~~Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~ e TSTF 65

NEW INSERT 1

Any deviation from the above guidelines manager shall be authorized in advance by the [Plant Superintendent] the plant managers or his designee, in accordance with approved administrative procedures, ~~or by higher levels of management, in accordance with established procedures~~ and with documentation of the basis for granting the deviation. ① TSTF 258

NEW INSERT 2

~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned.~~ Routine deviation from the ~~above~~ working hour guidelines ~~is not~~ shall not be authorized. TSTF 258

~~OR~~

~~[The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).]~~ TSTF 258

6.3.1 ⓔ ~~f~~. The [~~Operations Manager or Assistant Operations Manager~~] ~~shall hold an SRO license.~~ ① TSTF-65 ②

INSERT 3

6.2.4.1 ~~g~~ ⓕ. ~~The Shift Technical Advisor (STA)~~ an individual shall provide advisory technical support to the ~~Shift Supervisor (SS)~~ Unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA~~ This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. TSTF 258

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INSERT 1

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

INSERT 2

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

INSERT 3

Superintendent Operations shall hold (or have previously held) a Senior Reactor Operator License for North Anna or a similar design Pressurized Water Reactor plant. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station.

CTS

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

6.3

~~[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]~~ (8)

6.3.1 "*" 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]~~. (1) ~~The staff not covered by [Regulatory Guide 1.8]~~ shall meet or exceed the minimum qualifications of ~~[Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff]~~. (1)

TSTF-258

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ANSI 3.1 (12/79 Draft) for comparable positions. Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase." The Superintendent-Radiological Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

SS, Assistant SS, Control Room Operator-Nuclear, and the individual providing advisory technical support to the unit operations shift crew,

10 CFR 55.59(c) and 55.31(a)(4)

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5.3.2 For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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5.0 ADMINISTRATIVE CONTROLS

6.8 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a, 6.8.1.b, 6.8.1.c a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

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

a e

6.8.1.l c. Quality assurance for effluent and environmental monitoring;

6.8.1.f d. Fire Protection Program implementation; and

NEW e. All programs specified in Specification 5.5.

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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

6.15 5.5.1 Offsite Dose Calculation Manual (ODCM)

1.17 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

1.17 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 Annual Radioactive Effluent Release Reports required by Specifications ~~[~~5.6.2~~]~~ and Specification 5.6.3~~]~~. ① ③

Licensee initiated changes to the ODCM:

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

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

2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

6.15.b b. Shall become effective after the approval of the ~~[P~~plant manager ~~Superintendent]~~; and TSTF-65

6.15.c c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the

(continued)

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5.5 Programs and Manuals

6.15.c 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

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

6.8.4.d 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.

6.8.4.e 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

(continued)

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

6.8.4.e be taken whenever the program limits are exceeded. The program shall include the following elements:

6.8.4.e.1 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;

6.8.4.e.2 b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, INSERT 1 conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~ TSTF 258

6.8.4.e.3 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;

6.8.4.e.4 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;

6.8.4.e.5 e. Insert 3 ~~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;~~ TSTF-308

6.8.4.e.6 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;

6.8.4.e.7 g. Limitations on the dose rate resulting ~~from radioactive material released in gaseous effluents~~ from the site to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ INSERT 2 TSTF 258

6.8.4.e.8 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)

Rev. 0

INSERT 1

ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

INSERT 2

shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and
2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

INSERT 3

Determination of cumulative 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

CTS

5.5 Programs and Manuals

6.8.4.e 5.5.4 Radioactive Effluent Controls Program (continued)

6.8.4.e.9 i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

6.8.4.e.10 j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

INSERT 1 TSTF 258

5.7.1 5.5.5 Component Cyclic or Transient Limit (3) (1)

This program provides controls to track the U FSAR, Section ~~[ ]~~ 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~ (7)

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

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

4.4.10.1.1 5.5.~~7~~ 6 Reactor Coolant Pump Flywheel Inspection Program (7)

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

INSERT 2 TSTF-237

INSERT 3 (19)

(continued)

Rev. 0

INSERT 1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

INSERT 2

once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels.

INSERT 3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program surveillance frequency.

CTS

5.5 Programs and Manuals (continued)

4.0.5
4.4.10.1.2

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

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: TSTF-279

4.0.5.b

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.

4.4.5.2

5.5.~~9~~ 8 **Steam Generator (SG) Tube Surveillance Program** 7

INSERT

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

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test Frequencies. TSTF-118

(continued)

Rev. 0

INSERT

This program provides the controls for the inservice inspection of steam generator tubes to ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of steam generators is based on a modification of Regulatory Guide 1.83, Revision 1. This program shall include:

5.5.8.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.8-1.

5.5.8.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations > 20%, and

   2. Tubes in those areas where experience has indicated potential problems.

INSERT (CONTINUED)

3. A tube inspection (pursuant to Specification 5.5.8.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 5.5.8.2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

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

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

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

INSERT (CONTINUED)

5.5.8.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.8-2 at 40 month intervals fall into category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A major steam line or feedwater line break.

INSERT (CONTINUED)

5.5.8.4 Acceptance Criteria

a. As used in this Specification:

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

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

3. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.8.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely around the U-bend to the top support.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy-current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.8-2.

INSERT (CONTINUED)

STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

TABLE 5.5.8-1

Minimum Number of Steam Generators To Be Inspected During Inservice Inspection

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspection	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the unit) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that 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 sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

INSERT (CONTINUED)

STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

TABLE 5.5.8-2

Steam Generator Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample.
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 5.6.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes.	N/A	N/A

S=3[N/n]% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

CTS

5.5 Programs and Manuals (continued)

6.8.4.c 5.5.10 9 **Secondary Water Chemistry Program** 7

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include: 4

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

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

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

d. Procedures for the recording and management of data;

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

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

4.7.7.1 5.5.11 10 **Ventilation Filter Testing Program (VFTP)** 7

4.7.8.1
4.9.12

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]. 3 L 1 3

in general conformance with

requirements of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of

ANSI 1975 March 1978

4.7.7.1.b.1
4.7.7.1.e

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%]. 1 3 1

1.0

Regulatory Positions C.5.a and C.5.c of

1975 ANSI

ESF Ventilation System	Flowrate
[MCR/ESGR EVS ECCS PREACS]	[1000 cfm]

1

INSERT

Nominal accident flow for a single train actuation 15

(continued)

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Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber.

CTS

5.5 Programs and Manuals

Regulatory Positions C.5.a and C.5.d of (3)

5.5.~~11~~ 10 **Ventilation Filter Testing Program (VFTP)** (continued) (7)

4.7.7.1.b.1
4.7.7.1.f

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < ~~[0.05]~~ 1.0% when tested in accordance with [Regulatory Guide 1.52, Revision 2, March 1978 and ~~ASME~~ ANSI N510-~~1989~~ 1975] at the system flowrate specified below ~~[± 10%]~~. (1) (1)

ESF Ventilation System		Flowrate
[MCR/ESGR EVS ECCS PREACS]	Nominal accident flow for a single train actuation	1000 cfm

(1) (15)

INSERT

4.7.7.1.b.2

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of [Regulatory Guide 1.52, Revision 2], March 1978, shows the methyl iodide penetration less than the value specified below when tested in accordance with ~~[ASTM D3803-1989]~~ at a temperature of ~~≤~~ [30°C] (86°F) and ~~greater than or equal to the~~ relative humidity specified below. (3) (1) (1) (1) TSTF 362

NEW

ESF Ventilation System	Penetration	RH	~~Face Velocity~~
[MCR/ESGR EVS ECCS PREACS]	[2.5% 5%]	[70% 70%]	

TSTF-362 (13) (1)

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

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~ (1)

4.7.7.1.d

d. Demonstrate for ~~each of the ESF systems~~ the ECCS PREACS that the pressure drop across the combined HEPA filters, the ~~prefilters~~ demister filter, and the charcoal adsorbers is less than ~~the value specified below~~ 5" watergauge when tested in accordance with [Regulatory Guide 1.52. (16) (1)

(continued)

Rev. 0

INSERT

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber.

CTS

5.5 Programs and Manuals

March 1978

ANSI

4.7.7.1.d 5.5.~~11~~ 10 Ventilation Filter Testing Program (VFTP) (continued) 7

Revision 2, and ASME N510-~~1989~~ 1975 at ~~the~~ a system flowrate ~~specified below [± 10%].~~ ≤ 39,200 cfm 1

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

16

INSERT

~~e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].~~ 1

ESF Ventilation System	Wattage
~~[ ]~~	~~[ ]~~

12

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

5.5.~~12~~ 11 Explosive Gas and Storage Tank Radioactivity Monitoring Program 7

New

Decay Tanks 1

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 or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]. 1 1 1 1

The program shall include: Decay Tanks

3.11.2.5 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 1

(continued)

Rev. 0

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e. Demonstrate for the MCR/ESGR EVS that the differential pressure across the MCR/ESGR EVS fans is greater than 4" water gauge when tested in accordance with ANSI N510-1975 at a system flowrate of 1000 cfm ±10%.

CTS

5.5 Programs and Manuals

5.5.~~12~~ 11 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) 7

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

3.11.2.6 b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~ each gas storage tank ~~and fed into the offgas treatment system~~ ~~]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~ an uncontrolled release of the tanks' contents ~~]~~; and 1 1

3.11.1.4 c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following ~~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 ion exchanger system ~~[Liquid Radwaste Treatment System]~~ is less than the amount that would result in concentrations greater ~~less~~ than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, excluding tritium, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. INSERT 3 1 18 3

New 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.

New 5.5.~~13~~ 12 Diesel Fuel Oil Testing Program 7

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

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

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

(continued)

Rev. 0

INSERT

:

1. Refueling Water Storage Tank;
2. Casing Cooling Storage Tank;
3. PG Water Storage Tank;
4. Boron Recovery Test Tank; and
5. Any Outside Temporary Tank.

CTS

5.5 Programs and Manuals

New 5.5.~~13~~ 12 Diesel Fuel Oil Testing Program (continued) 7

2. ~~a flash point and~~ kinematic viscosity within limits for ASTM 2D fuel oil, and water and sediment ≤ 0.05% 9 10

3. ~~a clear and bright appearance with proper color;~~

verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil.

b. ~~Other properties for ASTM 2D fuel oil are within limits~~ within 31 days following ~~sampling and~~ addition to storage tanks; ~~and~~ of the new fuel oil TSTF-106

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every ~~31~~ 92 days in accordance with ASTM D-2276, Method A-2 or A-3. and 11

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies. TSTF-118

New 5.5.~~14~~ 13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications. 7

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~~ require either of the following: 3

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

2. ~~a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.~~ a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59. TSTF 364

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 3

d. Proposed changes that meet the criteria of Specification 5.5.1~~4~~3b 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). 3 7

(continued)

CTS

5.5 Programs and Manuals (continued)

New 5.5.~~15~~ 14 **Safety Function Determination Program (SFDP)** 7

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:

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. TSTF-273

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), 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.

INSERT 1

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. TSTF-273

TSTF-52

INSERT 2

Rev. 0

INSERT 1

CTS

NEW When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

INSERT 2

5.5.15 Containment Leakage Rate Testing Program

4.6.1.2
4.6.1.3.a

a. A program shall establish 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.

4.6.1.2
NEW

b. The Peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is ~~50~~ 45 psig. (3)

3.6.1.2.a

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

d. Leakage Rate acceptance criteria are:

3.6.1.2.a

1. ~~Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;~~ (17)

Prior to entering a MODE where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

≤ 0.60 L_a for the Type B and Type C tests on a Maximum Path Basis and ≤ 0.75 L_a for Type A tests.

During operation where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

≤ 1.0 L_a for overall containment leakage rate and ≤ 0.60 L_a for the Type B and Type C tests on a Minimum Path Basis.

CTS

INSERT 2 (continued)

3.6.1.3.b

2. ~~Air lock~~ testing acceptance ~~criteria are~~ criterion is:

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

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

(14)

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

NEW f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

CTS

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

6.9.1.5.a 5.6.1 Occupational Radiation Exposure Report ①

------------------------------NOTE------------------------------
6.9.1.5.a footnote 1 A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

INSERT

~~A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.]~~ TSTF-152

6.9.1.8 5.6.2 Annual Radiological Environmental Operating Report ①

------------------------------NOTE------------------------------
6.9.1.8 footnote * 6.9.1.5 footnote 1 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 ③

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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 (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, 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.

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5.6 Reporting Requirements

6.9.1.8 5.6.2 Annual Radiological Environmental Operating Report (continued)

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

New

commensurate with the format in the ODCM.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]~~. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. (1) TSTF-348

Annual

6.9.1.9 5.6.3 Radioactive Effluent Release Report (3) TSTF-152

6.9.1.9 footnote *

shall

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal ~~should~~ combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. (1)
----- (3)

Annual

in the previous year

prior to May 1 of each year

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1. TSTF-152 TSTF-152

(continued)

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5.6 Reporting Requirements (continued)

6.9.1.6 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. TSTF 258

6.9.1.7 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:

INSERT 1

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

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

INSERT 2

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ and TSTF-363 ①

~~The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).~~

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. TSTF-363

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

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ ⑤

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1. Safety Limits,
2. Shutdown Margin,
3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. Axial Flux Difference limits,
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
10. RCS Pressure, Temperature, and Flow DNB Limits, and
11. Boron Concentration.

INSERT 2

1. VEP-FRD-42, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL – 1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS – SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."
6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."
7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY-REFERENCE CORE REPORT."
8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."
9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."
10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

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5.6 Reporting Requirements (continued)

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

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

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

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

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

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

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

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

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

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

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance

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5.6 Reporting Requirements

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with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

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

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

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

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement. TSTF-37

New 5.6.~~8~~6 PAM Report TSTF-37

When a report is required by Condition B ~~or G~~ of LCO 3.3.~~[3]~~ f g, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. 1 3

(continued)

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5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken. 7

6.9.1.5.b 5.6.10 Steam Generator Tube Inspector Report TSTF-37
4.4.5.5 8 7

Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used. 1 7

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Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications. 8

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5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

    1. Number and extent of tubes inspected.

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

    3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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5.0 ADMINISTRATIVE CONTROLS

6.12 {5.7 High Radiation Area}

6.12.1 5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection ~~{Health Physics}~~ Technicians}) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. TSTF 65

6.12.1 Note

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 Health Physicist ~~[Radiation Protection Manager]~~ in the RWP. TSTF 65

6.12.2 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 Foreman 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

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5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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[5.7 High Radiation Area]

[5.7.2 (continued)

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.]

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1. The brackets are removed and the proper plant specific information/value is provided.

2. The statement in ISTS 5.2.2.f is modified to state, "The Superintendent Operations shall hold (or have previously held) a Senior Reactor Operator License for North Anna or a similar design Pressurized Water Reactor plant. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station." This is consistent with the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. Reference to low pressure turbine disc stress corrosion cracking associated with the secondary water chemistry program is deleted because it is not applicable to NAPS. There has been no evidence of low pressure turbine disc stress corrosion cracking at NAPS. EPRI secondary water chemistry guidelines do not note any relation between secondary water chemistry and low pressure turbine disc stress corrosion cracking. This is consistent with the current licensing basis.

5. ISTS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," is not adopted in the ITS. CTS Figures 3.4-2 and 3.4-3, which provide Reactor Coolant System heatup and cooldown limitations, respectively, were adopted in ITS Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits." Subsequent Specifications are renumbered accordingly.

6. ITS 5.2.2.a is modified to require four non-licensed operators be assigned for each control room from which a reactor is operating in MODE 1, 2, 3 or 4. This is based on preference to support plant assumptions regarding available non-licensed operators and is consistent with the current licensing basis. The non-licensed operator assigned to each unit containing fuel may be one of these four assigned to the control room.

7. The ISTS 5.5.6 requirement, "Pre-Stressed Concrete Containment Tendon Surveillance Program," is not adopted because it is not applicable to the North Anna design. The ISTS 5.6.9 requirement, "Tendon Surveillance Report," is also not adopted. The containment at North Anna is a steel-lined, heavily reinforced concrete structure with vertical cylindrical wall and hemispherical dome, supported on a flat base mat. Subsequent Specifications are renumbered accordingly.

8. The information contained in the reviewer's note is not retained.

9. The ISTS 5.5.13.a.2 flash point test requirement for determining acceptability of new fuel oil for use prior to addition to the storage tanks is not adopted. This test will be conducted as part of testing to be completed within 31 days following addition of the new fuel oil to the storage tanks. Flash point determination of new fuel oil is not currently

performed, and has not been found to be essential to provide assurance that the new fuel oil is acceptable prior to addition of new fuel oil to the storage tanks.

10. The ISTS 5.5.13.a.3 requirement to determine a clear and bright appearance with proper color as part of determining acceptability of new fuel oil prior to addition to the storage tanks is not adopted, and a test for water and sediment being ≤ 0.05 percent is adopted instead. The water and sediment test is adopted because the diesel fuel oil is dyed.

11. The ISTS 5.5.13.c requirement to determine, "Total particulate concentration of the fuel oil" every 31 days is modified. ITS 5.5.12.c adds the word "stored" in front of the term "fuel oil" to clarify that the test is to be performed on stored fuel oil rather than new fuel oil. The frequency of the test is changed from 31 days to 92 days based on plant operating practice of conducting the test every 92 days, test history indicating that the interval is appropriate, and there being no current Technical Specification requirement to perform the test.

12. The ISTS 5.5.11.e bracketed requirement to demonstrate ESF systems ventilation filter heater heat dissipation capability is not adopted. The ESF systems ventilation systems heaters at NAPS are not required for Operability of the ventilation systems, they are only required for performance of the surveillance test. A separate test in the Technical Specifications is not warranted and is consistent with the current licensing basis.

13. Face velocity is not adopted as one of the required parameters for testing charcoal adsorbers in ISTS 5.5.11.c. The system does not have a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), and according to TSTF-362 is thus not required to be specified in the ITS.

14. ISTS 5.5.15 Containment Leakage Rate Testing Program air lock testing acceptance criterion d.2.b) is not adopted. ISTS 5.5.15.d.2.b) states, "For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig." North Anna uses criterion 5.5.15.d.2.a), which states, "Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a." ISTS 5.5.15.d.2.a) provides an acceptable leakage rate criterion for the air lock doors, and ISTS 5.5.15.d.2.b) is not required.

15. An explanation is added to ISTS 5.5.11.a and ISTS 5.5.11.b for the phrase, "Nominal accident flow for a single train actuation," which is used for the ECCS PREACS flowrate designated. Use of nominal accident flow is a better measure than a specific flow value of whether the filters will perform their function, since this is the flow that will occur in case of a DBA. This explanation is consistent with current licensing basis and plant design.

16. ISTS 5.5.11.d is modified to reflect that the criteria for pressure drop across the combined HEPA filters, the demister filter, and the charcoal adsorbers, apply to only one system, ECCS PREACS, using a maximum flowrate. ISTS 5.5.11.d is also modified to reflect that the criteria for the MCR/ESGR EVS is for pressure differential across the

MCR/ESGR fans. This is consistent with the physical arrangement of the equipment and the licensing basis at NAPS. The testing criteria still demonstrate proper system operation.

17. STS 5.5.15.d.1 is modified to specifically address containment leakage rate requirements prior to entering a MODE where containment OPERABILITY is required, and during operation where containment OPERABILITY is required. The requirements adopted in ITS 5.5.15.d.1 are consistent with the CTS requirements, and encompass the requirements of ISTS 5.5.15.d.1.

18. ISTS 5.5.12.c is modified to clarify that the surveillance program described limits the radioactivity contained in the specified outdoor liquid radwaste tanks to less than the amount that would result in concentrations greater than, rather than less than, the limits of 10 CFR 20, Appendix B, Table 2, Column 2 in case of the specified event. ISTS 5.5.12.c is also modified to clarify that the radioactivity limits exclude limits on tritium. These changes are consistent with the current licensing basis and guidance in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," section 4.4.

19. ISTS 5.5.7 is modified to state that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program surveillance frequency. This allowance is consistent with the current licensing basis, and is consistent with the NUREG-1431 format of retaining these allowances for other current Technical Specification requirements that have been moved to Section 5.0.

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS BASES

MARKUP AND JUSTIFICATION FOR DEVIATIONS

Chapter 5.0 does not have Bases

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

CURRENT TECHNICAL SPECIFICATIONS

MARKUP AND DISCUSSION OF CHANGES

UNIT 1

A.1

ITS 5.0

ITS

06-23-98

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

5.1.1 6.1.1 The ~~Site Vice President~~ plant manager shall be responsible for overall facility operation. ~~In his absence, the Manager - Station Operations and Maintenance shall be responsible for overall facility operation.~~ During ~~the~~ his absence ~~of both~~, the ~~Site Vice President~~ shall delegate in writing the succession to this responsibility. ← INSERT 1 L.4 M.1

5.1.2 6.1.2 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function and shall be the only individual that may direct the licensed activities of licensed operators. ~~A management directive to this effect, signed by the Senior Vice President - Nuclear, shall be reissued to all station personnel on an annual basis.~~ L.5

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

5.2.1 6.2.1 Onsite and Offsite Organization

An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1 a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the UFSAR /QA Plan. INSERT 2 M.7

5.2.1.b b. The ~~Site Vice President~~ plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. L.6

5.2.1.c c. ~~The Vice President - Nuclear Operations~~ A specified corporate officer 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. L.6

5.2.1.d d. The ~~management position~~ individuals responsible for training of the operating staff and ~~the management position responsible~~ for the quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom including sufficient independence from ~~cost and schedule when opposed to safety considerations~~ operating pressures. M.15

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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.

INSERT 2

including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications

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may report to the appropriate onsite manager, however, these individuals

individuals

sufficient organizational freedom to ensure their independence from operating pressures

ADMINISTRATIVE CONTROLS

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5.2.1.d e. The management position responsible for health physics ~~shall have direct access to that onsite individual having responsibility for overall facility management. Health physics personnel~~ shall have the authority to cease any work activity when worker ~~safety is jeopardized or~~ in the event of unnecessary personnel radiation exposures.

FACILITY STAFF

include

5.2.2 6.2.2 The Facility organization shall be as shown in the UFSAR. L.3

5.2.2.a a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

~~b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODES 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.~~ A.34

5.2.2.c radiation protection c. A health physics technician# shall be onsite when fuel is in the reactor. L.11

~~d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~

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ITS 5.2.2.c # The health physics technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions. L.11

~~6.2.3 STATION NUCLEAR SAFETY (SNS)~~

~~FUNCTION~~

~~6.2.3.1 SNS shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources which may indicate areas for improving plant safety.~~

~~COMPOSITION~~

~~6.2.3.2 SNS shall be composed of at least five dedicated, full-time engineers located onsite.~~ LA.4

~~RESPONSIBILITIES~~

~~6.2.3.3 SNS shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.~~

~~6.2.3.4 SNS shall disseminate relevant operational experience.~~

~~AUTHORITY~~

~~6.2.3.5 SNS shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Manager - Station Safety and Licensing.~~

6.2.4 SHIFT TECHNICAL ADVISOR An individual Unit operations shift crew A.28

5.2.2.f 6.2.4.1 ~~The Shift Technical Advisor~~ shall serve in an advisory capacity to ~~Shift Supervisor~~ on ~~matters pertaining to the engineering aspects of assuring~~ safe operation of the unit.

the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the

This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. M.12

~~* Not responsible for sign-off function~~ LA.4

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TABLE 6.2-1[a]

MINIMUM SHIFT CREW COMPOSITION

Total Staffing Requirements for Station Operation

With Either or Both Units in Mode 1, 2, 3 or 4 A.2

POSITION - NUMBER - CONDITIONS

POSITION		NUMBER	CONDITIONS
SS	-	ONE	(Shift Supervisor may fulfill duties for both units).
SRO	-	ONE	(If ONE unit is in MODE 5, 6 OR DEFUELED, Senior Reactor Operator is assigned to the Unit in MODE 1, 2, 3 or 4). L.8
RO	-	THREE	(ONE Reactor Operator is assigned to each unit PLUS one is shared by both units).
5.2.2.a AO	-	FOUR	(TWO Auxiliary Operators are assigned to each unit).
STA	-	ONE	(Shift Technical Advisor may fulfill duties for both units).

L.9

With Both Units in Mode 5 or 6 (or ~~DEFUELED~~) A.2

POSITION - NUMBER - CONDITIONS

POSITION		NUMBER	CONDITIONS
SS	-	ONE	(Shift Supervisor may fulfill duties for both units).
SRO	-	NONE	
RO	-	TWO	(ONE Reactor Operator is assigned to each unit). L.8
5.2.2.a AO	-	~~TWO~~ One	(ONE Auxiliary Operator is assigned to each unit). L.9
STA	-	ONE	(Shift Technical Advisor may fulfill duties for both units). A.2

a - This Table and Table 6.2.1 of Unit 2 Technical Specifications represent Total Station Staffing and ARE NOT ADDITIVE. A.2

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TABLE 6.2-1 (Continued)

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1. (A.12)
SRO - Individual with a Senior Reactor Operators License on Unit 1.
RO - Individual with a Reactor Operators License on Unit 1.
AO - Auxiliary Operator
STA - Shift Technical Advisor

10 CFR 50.54(m)(2)(ii) and 5.2.2.a and 5.2.2.f (A.2)

ITS 5.2.2.b Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 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 of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. (L.1) (A.2) (L.1)

ITS 5.1.2 During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command function. (L.18) (A.6) (L.18)

limit (L.24)

ITS 5.2.2.d Procedures will be established to insure that NRC policy statement guidelines regarding work hours established for employees are followed. In addition, procedures will provide for documentation of authorized deviations from these guidelines and that the documentation is available for NRC review. (L.10)

Insert 5.2.2.d

(M.18)

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6.3 FACILITY STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1 (12/79 Draft)* for comparable positions, except for:

5.3.1 1. The Superintendent - Radiological Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.1 2. Incumbents in the positions of Shift Supervisor, Assistant Shift Supervisor (SRO), Control Room Operator - Nuclear (RO), and ~~Shift Technical Advisor~~, shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). INSERT A.29

the individual providing advisory technical support to the unit operations shift crew A.28

5.2.2.e 3. The Superintendent Operations shall hold (or have previously held) a Senior Reactor Operator License for North Anna Power Station or a similar design Pressurized Water Reactor plant.

5.2.2.e 4. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station.

~~6.4 TRAINING~~

~~6.4.1 The Manager - Nuclear Training is responsible for ensuring that retraining and replacement training programs for the licensed facility staff meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). Also, a retraining and replacement training program for non-licensed facility staff shall meet or exceed the recommendations of Section 5 of ANS 3.1 (12/79 Draft)*.~~ L.19

~~6.5 REVIEW AND AUDIT~~

~~6.5.1 STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC)~~ LA.6

~~FUNCTION~~

~~6.5.1.1 The SNSOC shall function to advise the Site Vice President on all matters related to nuclear safety.~~

~~* Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase."~~ L.19

INSERT

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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COMPOSITION

6.5.1.2 The SNSOC shall be composed of:

Chairman:	Manager - Station Safety and Licensing
Vice Chairman and Member:	Manager - Station Operations and Maintenance
Member:	Superintendent - Operations
Member:	Superintendent - Maintenance
Member:	Superintendent - Radiological Protection
Member:	Superintendent - Engineering

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SNSOC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in SNSOC activities at any one time.

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MEETING FREQUENCY

6.5.1.4 The SNSOC shall meet at least once per calendar month and as convened by the SNSOC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all new procedures required by Specifications 6.8.1 and 6.8.2, 2) all procedure changes that require a safety evaluation, 3) all programs required by Specification 6.8.4 and changes thereto, and 4) any other procedures or changes thereto as determined by the Site Vice President to affect nuclear safety. LA.6

b. Review of all proposed tests and experiments that affect nuclear safety.

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

d. Review of all proposed changes to Appendix "A" Technical Specifications and Appendix "B" Environmental Protection Plan. Recommended changes shall be submitted to the Site Vice President.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Operations and the MSRC.

f. Review of all REPORTABLE EVENTS and Special Reports.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee or Site Vice President.

i. Deleted.

j. Deleted.

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k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Operations and the Management Safety Review Committee.

l. Review changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

m. Review of the Fire Protection Program and implementing procedures and shall submit recommended changes to the Site Vice President.

AUTHORITY

6.5.1.7 The SNSOC shall:

a. Provide written approval or disapproval of items considered under 6.5.1.6(a) through (c) above. SNSOC approval shall be certified in writing by either the Manager - Station Operations and Maintenance or the Manager - Station Safety and Licensing.

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

c. Provide written notification within 24 hours to the Vice President- Nuclear Operations and the Management Safety Review Committee (MSRC) of disagreement between the SNSOC and the Site Vice President; however, the Site Vice President shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

LA.6

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Site Vice President, Vice President-Nuclear Operations and the MSRC.

6.5.2 MANAGEMENT SAFETY REVIEW COMMITTEE (MSRC)

FUNCTION

6.5.2.1 The MSRC shall function to provide independent review of designated activities in the areas of:

a. Station Operations
b. Maintenance
c. Reactivity Management
d. Engineering
e. Chemistry and Radiochemistry
f. Radiological Safety
g. Quality Assurance Practices
h. Emergency Preparedness

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COMPOSITION

6.5.2.2 The MSRC shall be composed of the MSRC Chairman and a minimum of four MSRC members. The Chairman and all members of the MSRC shall have qualifications that meet the requirements of Section 4.7 of ANSI/ANS 3.1-1979 Rev. 1 (Draft).

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the MSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in MSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants should be utilized as determined by the MSRC Chairman to provide expert advice to the MSRC.

MEETING FREQUENCY

6.5.2.5 The MSRC shall meet at least once per calendar quarter.

QUORUM

6.5.2.6 The minimum quorum of the MSRC necessary for the performance of the MSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 50% of the MSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The MSRC shall be responsible for the review of:

a. Safety evaluations as programmatically discussed in the Updated Final Safety Analysis Report for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to assess the effectiveness of the safety evaluation program and to verify that the reviewed actions did not constitute an unreviewed safety question.

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

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

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

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

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

g. Events requiring written notification to the Commission.

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

i. A representative sample of reports and meetings minutes of the SNSOC.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the MSRC. These audits shall encompass:

LA.6

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.

b. The performance, training and qualifications of the entire facility staff.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50.

e. Any other area of facility operation considered appropriate by the MSRC or the Vice President - Nuclear Operations.

f. The Fire Protection Program and implementing procedures.

g. An independent fire protection and loss prevention inspection and audit shall be performed utilizing an outside qualified fire consultant.

h. The Radiological Environmental Monitoring Program and the results thereof.

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i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes.

AUTHORITY

6.5.2.9 The MSRC shall report to and advise the Senior Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

(LA.6)

RECORDS

6.5.2.10 Records of MSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each MSRC meeting shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days of each meeting.

b. Reports of reviews with safety significant findings encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

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6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS: A.19

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Vice President-Nuclear Operations and the MSRC. LA.6

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President-Nuclear Operations and MSRC shall be notified within 24 hours. See ITS 2.0

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President-Nuclear Operations and the MSRC within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

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

b. Refueling operations. A.3

Insert proposed ITS 5.4.1.b M.2

Insert proposed ITS 5.4.1.e M.3

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5.5.1 6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

5.5.1.a (2nd) a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.r. This documentation shall contain: A.11

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

5.5.1.a.2 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b (2nd) b. Shall become effective after review and acceptance by the SNSOC and the approval of the ~~Site Vice President~~ plant manager. LA.6 L.6

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

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OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a 1.17 The OFFSITE DOSE CALCULATION MANUAL (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/trip setpoints, and
5.5.1.b in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the ~~Radioactive Effluent~~ Controls and Radiological Environmental Monitoring
activities ~~Programs required by Section 6.8.4~~ and (2) descriptions of the information that should be ~~included in the~~ Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications ~~6.9.1.8~~ 5.6.2 and ~~6.9.1.9~~ 5.6.3. A.20

OPERABLE - OPERABILITY

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

see ITS 1.0

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

~~PROCESS CONTROL PROGRAM~~

~~1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.~~ L.32

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. see ITS 1.0

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c. ~~Surveillance and test activities of safety related equipment.~~ (A.3)

d. ~~Security Plan implementation.~~ (A.4)

e. ~~Emergency Plan implementation.~~

5.4.1.d f. Fire Protection Program implementation.

g. ~~PROCESS CONTROL PROGRAM implementation.~~ (L.32)

5.5.1 h. OFFSITE DOSE CALCULATION MANUAL implementation.

5.4.1.c i. Quality Assurance Program for effluent and environmental monitoring, ~~using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.~~ (LA.1) (L.30)

~~6.8.2 Each new procedure of 6.8.1 above, except 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved by the SNSOC prior to implementation as set forth in administrative procedures.~~ (LA.6)
~~Procedures of 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved as set forth in the facility's Security Plan, Emergency Plan, and section 6.5.1.6.m of the Technical Specifications, respectively.~~ (L.30)

~~6.8.3 Procedure changes that require a safety evaluation shall also be reviewed and approved by SNSOC. All other changes shall be independently reviewed and approved as programmatically discussed in the Updated Final Safety Analysis Report.~~ (LA.6) (L.30)

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

5.5.2 a. Primary Coolant Sources Outside Containment

A program ~~to reduce~~ 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 as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following: (L.12)

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

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b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following: (LA.3)

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

5.5.9 c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all control off point chemistry conditions, and (A.3S)

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

5.5.3 d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

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5.5.4 e. Radioactive Effluent Controls Program

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

5.5.4.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.4.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times 10 CFR ~~Part 20~~, 20.1001 - 20.2402 Appendix B, Table 2, Column 2, (A.30)

5.5.4.c 3) 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,

5.5.4.d 4) 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 Appendix I to 10 CFR Part 50,

5.5.4.e 5) 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, (L.31)

5.5.4.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.4.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

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

b) For Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr. to any organ.

Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

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5.5.4.h 8) 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 Appendix I to 10 CFR Part 50,

5.5.4.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR 50,

5.5.4.j 10) 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 Part 190. L.25

INSERT

~~f. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radio nuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,~~ LA.10

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

~~g. Configuration Risk Management Program~~

~~The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:~~ LA.8

~~1) Provisions for the control and implementation of a Level 1, at power, internal events, PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.~~

~~2) Provisions for performing an assessment prior to entering the LCO Action Statement for planned activities.~~

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The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

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Configuration Risk Management Program (continued)

3) Provisions for performing an assessment after entering the LCO Action Statement for unplanned entry into the LCO Action Statement.

4) Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action Statement.

5) Provisions for considering other applicable risk significant contributors such as Level 2 issue and external events, qualitatively or quantitatively.

Current risk-informed action statements include: Action 3.8.1.1.b; 3.4.3.2.A.2; 3.3.1.1: 3.3.2.1

LA.8

6.9 REPORTING REQUIREMENTS

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

ROUTINE REPORTS

5.6 6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted. A.24

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report. L.7

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ITS

DESIGN FEATURES

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet. Mean Sea Level, USGS datum.

See ITS 4.3.2

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.

See ITS 4.3.3

5.5.5 5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

the UFSAR, Section 5.2,

5.7.1 The components identified in ~~Table 5.7-1~~ are designed and shall be maintained within the cyclic or transient design limits ~~of Table 5.7-1~~.

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at 100°F/hr and 200 cooldown cycles at 100°F/hr	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at 200°F/hr	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER. (Full Power Trip)
	10 inadvertent pressurizer auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.

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ITS

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

5.5.6 4.4.10.1.1 In addition to the requirements of Specification 4.0.5, the Reactor Coolant pump flywheels shall be inspected once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels. A.22

5.5.7 4.4.10.1.2 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval. LA.11

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ITS 5.0

ITS 5.5 5.5.7

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

See ITS 3.0

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

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition sh[all] not be made unless the Surveillance Requirement(s) associated with the Limiting Condition fo[r] Operation have been performed within the stated surveillance interval or as otherwise specified.

5.5.7 4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

A.21

5.5.7.a a. ~~Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).~~

ITS

5.5 APPLICABILITY

5.5.7 SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.a b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: A.21

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and 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

13

A.13

5.5.7.b c. The provisions of Specification ~~4.0.2~~ SR3.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. A.21 A.20

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d. ~~Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.32

5.5.7.d e Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification. L.20

Insert proposed ITS 5.5.7.c

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5.5.9 Steam Generator (SG) Tube Surveillance Program

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ITS

REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

~~3.4.5 Each steam generator in a non-isolated reactor coolant loop shall be OPERABLE.~~

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

~~ACTION:~~

~~With one or more steam generators in non-isolated reactor coolant loops inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

See ITS 3.4.13

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the required Specification 4.0.5.~~

A.7

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies

5.5.8.1 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table ~~4.4-1~~ 5.5.8-1.

5.5.8.2 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.4-2~~ 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification ~~4.4.5.3~~ 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification ~~4.4.5.4~~ 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

5.5.8.2.b b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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5.5.8.2.b.1 1. All nonplugged tubes that previously had detectable wall penetrations >20%, and

5.5.8.2.b.2 2. Tubes in those areas where experience has indicated potential problems.

5.5.8.2.b.3 3. A tube inspection (pursuant to Specification ~~4.4.5.4~~ 5.5.8.4 a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection. A.20

5.5.8.2.c c. The tubes selected as the second and third samples (if required by Table ~~4.4-2~~ 5.5.8-2) during each inservice inspection may be subjected to a partial tube inspection provided: A.20

5.5.8.2.c.1 1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

5.5.8.2.c.2 2. The inspections include those portions of the tubes where imperfections were previously found.

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

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

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

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REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.3 4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

5.5.8.3.a a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

5.5.8.3.b b. If the results of the inservice inspection of a steam generator conducted in accordance with Table ~~4.4-2~~ 5.5.8-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification ~~4.4.5.3.a~~ 5.5.8.3.a; the interval may then be extended to a maximum of once per 40 months. A.20

5.5.8.3.c c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.4-2~~ 5.5.8-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification ~~3.4.6.2~~ 3.4.13.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A major steam line or feedwater line break.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.4 4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in ~~4.4.5.3.c~~ 5.5.8.3.c, above. (A.20)

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely around the U-bend to the top support.

ITS

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENT (Continued)

5.5.8.4

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2 5.5.8-2

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4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of an imperfection.
3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

5.5.8-1

TABLE ~~4.4-1~~

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that 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 sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

Table 5.5.8-2

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Report to NRC ~~& obtain approval prior to operation~~ pursuant to 5.6.7.c	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit and n is the number of steam generators inspected during an inspection

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE: See ITS 3.7.10

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by: LA.5

5.5.10.a 5.5.10.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e and f.). LA.5

5.5.10.c 2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.10.a 5.5.10.b 3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

5.5.10.c c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%. LA.5

d. At least once per 18 months by: LA.5

5.5.10.e 1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

INSERT

Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at frequencies in general conformance with, and in accordance with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of, Regulatory Guide 1.52, Revision 2, and ANSI N510-1975.

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

PLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

ITS

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of ≥ 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

5.5.10.a e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.S

5.5.10.b f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.S

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 2) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of ≥ 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is ≤ 120°F.

See ITS 3.7.11

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PLANT SYSTEMS

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two safeguards area ventilation systems (SAVS) shall be OPERABLE with:

a. one SAVS exhaust fan

b. one auxiliary building HEPA filter and charcoal adsorber assembly (shared with Unit 2)

See ITS 3.7.12

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one SAVS inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SAVS system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Initiating, from the control room, flow through the auxiliary building HEPA filter and charcoal adsorber assembly and verifying that the SAVS operates for at least 10 hours with the heater on.

b. ~~At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system,~~ by: LA.S

5.5.10.a+b

1. Verifying that ~~the~~ ECCS cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ~~6,300 cfm ± 10%~~ nominal accident flow for a single train actuation ~~(except as shown in Specifications 4.7.8.1e. and f.)~~. M.21 10 LA.S M.21

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Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber.

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SURVEILLANCE REQUIREMENTS (Cont'd)

5.5.10.c 2. Verif~~ying, within 31 days after removal,~~ that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~ 70%. LA.S L.33

5.5.10.a 5.5.10.b 3. Verifying ~~a system~~ flow rate ~~of 6,300 cfm ± 10% during operation~~ of one ECCS PREACS train provides greater than the minimum required cooling flow for ECCS equipment when tested in accordance with ANSI N510-1975. M.21

5.5.10.c c. ~~Within 31 days of completing 720 hours of charcoal adsorber operation,~~ verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~ 70%. LA.S L.33

d. ~~At least once per 18 months by:~~ LA.S

5.5.10.d 1. Verifying that the ~~pressure drop~~ across the HEPA filter and charcoal adsorber assembly is < ~~6~~ 5 inches Water Gauge while operating the ~~ventilation system~~ ECCS PREACS at a flow rate of ~~6,300 cfm ± 10%~~ ≤ 39,200 cfm. M.21

2. Verifying that on a Containment Hi-Hi Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly. See ITS 3.7.12

5.5.10.a e. ~~After each complete or partial replacement of a HEPA filter bank by~~ verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. LA.S M.21

5.5.10.b f. ~~After each complete or partial replacement of a charcoal adsorber bank by~~ verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. LA.S

one ECCS PREACS train at nominal accident flow M.21

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~~Specifications 3/4.11.1.1 through 3/4.11.1.3 have been deleted~~

ITS 5.0

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~~Specifications 3/4.11.2.1 through 3/4.11.2.4 have been deleted~~

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RADIOACTIVE STORAGE

3/4.11.2 GAS STORAGE

EXPLOSIVE GAS MIXTURE

LIMITING CONDITIONS FOR OPERATION

5.5.11.a 3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration could exceed 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the affected waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the affected waste gas decay tank greater than 4% volume immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 4% by volume without delay, then continue with Action "a" above.

c. With the requirements of Action "a" not satisfied, immediately suspend all additions of waste gases to the affected tank until the oxygen concentration is restored to less than 2% by volume and submit a Special Report to the commission pursuant to Specification 6.9.2 within the next 30 days outlining the following:

1. The cause of the waste gas decay tank exceeding the 2% oxygen limit,
2. the reason why the oxygen concentration could not be returned to within limits, and
3. actions taken and the time required to return the oxygen concentration to within limits.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

LA.7

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be determined to be within the above limits by continuously monitoring the waste gases in the inservice waste gas decay tank with the oxygen monitor required OPERABLE by Table 3.3-14 of Specification 3.3.3.11.

INSERT proposed 5.5.11.a

M.11

INSERT

Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Decay Tanks, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The Liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Release due to Tank Failure."

(A.1)

ITS 5.0

ITS

7-19-90

RADIOACTIVE STORAGE

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

5.5.11.b 3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ≤ 25,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

LA.7

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is ≤ 1.0 µCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 µCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.

Insert proposed 5.5.11.b

M.11

A.1

ITS

RADIOACTIVE STORAGE

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

5.5.11.c 3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. LA.7

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank** LA.7

APPLICABILITY: At all times. LA.7

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per week when radioactive materials are being added to the tank.

INSERT Proposed 5.5.11.c M.11

~~*This is a shared system with Unit 2.~~ LA.7

**Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste ion exchanger system.

INSERT → A.18

INSERT

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.

A.1

~~Specifications 3/4.11.3 through 3/4.11.4 have been deleted~~

Insert proposed 5.5.12

M.5

Insert proposed 5.5.13

M.6

Insert proposed 5.5.14

L.14

Insert proposed 5.5.15

A.10

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.1

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves, secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

See ITS 3.6.1

ITS 5.5.15

c. ~~After each closing of the equipment hatch, by leak rate testing the equipment hatch seals, with gas at P_a, greater than or equal to 44.1 psig. Results shall be evaluated against the criteria of Specification 3.6.1.2.b as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995.~~

d. ~~Each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line.~~

A.26

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked sealed or otherwise sealed in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such surveillance need not be performed more often than once per 92 days.

See ITS 3.6.3

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

5.5.15 3.6.1.2 Containment leakage rates shall be limited to:

5.5.15.c 5.5.15.d.1 5.5.15.b

a. An overall integrated leakage rate of less than or equal to L_a, 0.1 percent by weight of the containment air per 24 hours, at the calculated peak containment pressure P_a, greater than or equal to 44.1 psig. The containment design pressure is 45psig. M.20

5.5.15.d.1 5.5.15.b

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, greater than or equal to 44.1 psig.

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

ACTION:

5.5.15.d.1 With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

See ITS 3.6.1 / See ITS 3.6.1

SURVEILLANCE REQUIREMENTS

5.5.15.a 4.6.1.2 The containment and containment penetrations shall be tested by performing leakage rate testing as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. ~~The provisions of Specification 4.0.2 are not applicable.~~ A.16

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

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program. A.36

ITS 5.0

A.1

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A.1

ITS 5.0

ITS

02-09-96

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with: ⟨See ITS 3.6.2⟩

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

5.5.15.b 5.5.15.d.2 b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a greater than or equal to 44.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: ⟨See ITS 3.6.2⟩

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.+

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With a containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

5.5.15.a 4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By performing leakage rate testing as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. ~~The provisions of~~

5.5.15.f ~~Specification 4.0.2 are not applicable.~~

b. At least once per refueling outage by verifying that only one door in each air lock can be opened at a time. ⟨See ITS 3.6.2⟩

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

A.16

+ Entry to repair the inner air lock door, if inoperable, is allowed. ⟨See ITS 3.6.2⟩

(A.1)

ITS 5.0

2-17-94

ITS

ANNUAL REPORTS[1] for the Steam Generator Tube Inspection Report and by April 30 of each year for the Occupational Radiation Exposure Report (L.15)

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. ~~The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ (A.25)

collective deep dose equivalent (reported in person-rem)

6.9.1.5 Reports ~~required on an annual basis shall include:~~

for whom monitoring was performed / an annual deep dose equivalent

5.6.1 a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions. (M.14) (L.21) (M.14)

electronic dosimeter

deep dose equivalent

5.6.7 b. The complete results of the steam generator tube ~~inservice inspections~~ performed during the report period (Reference Specification ~~4.4.5.5.b.~~) 5.6.7 (A.20)

~~c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ (L.2)

Note: 5.6.1, 5.6.2, 5.6.3 1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

5.6.1 2/ This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20.

5.6.2 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

INSERT →

M.8

Note 5.6.2 * A single submittal may be made for a multiple unit station.

INSERT

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3

ANNUAL RADIOLOGICAL EFFLUENT RELEASE REPORT*

6.9.1.9 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. 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 (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

shall (M.10)

Note 5.6.3 * A single submittal may be made for a multiple unit station. The submittal ~~should~~ combine those sections that are 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.

(A.1)

ITS

ADMINISTRATIVE CONTROLS

L.26

5.6.4 MONTHLY OPERATING REPORT

~~6.9.1.6~~ Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

A.14

5.6.5 CORE OPERATING LIMITS REPORT

5.6.5.a 6.9.1.7.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Safety Limits,
2. Shutdown Margin,

1. Moderator Temperature Coefficient ~~BOC and EOC limits, and 300 ppm and 60 ppm surveillance limits for Specification 3/4.1.1.4,~~

2. Shutdown Bank Insertion Limit ~~for Specification 3/4.1.3.5,~~

3. Control Bank Insertion Limits ~~for Specification 3/4.1.3.6,~~

4. Axial Flux Difference limits ~~for Specification 3/4.2.1,~~

5. Heat Flux Hot Channel Factor, ~~K(Z), N(Z) for Specification 3/4.2.2, and~~

6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier, ~~for Specification 3/4.2.3.~~

9. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
10. RCS Pressure, Temperature, and Flow DNB limits, and
11. Boron Concentration

A.37

5.6.5.b 6.9.1.7.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC ~~as identified in 6.9.1.7.e~~, specifically those described in the following documents:

5.6.5.c 6.9.1.7.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d 6.9.1.7.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A.14

6.9.1.7.e REFERENCES

5.6.5.b

1. VEP-FRD-42~~, Rev. 1-A,~~ "Reload Nuclear Design Methodology," ~~September 1986~~. LA.9

~~(Methodology for LCO 3.1.1.4 - Moderator Temperature Coefficient, LCO 3.1.3.5 - Shutdown Bank Insertion Limit, LCO 3.1.3.6 - Control Bank Insertion Limits, LCO 3.2.2 - Heat Flux Hot Channel Factor, LCO 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~ LA.9

5.6.5.b

2a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary). LA.9

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2b. 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 IN W ECCS EVALUATION MODEL", JULY 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2f. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY REPORT," June 1990 (W Proprietary). REFERENCE CORE A.27

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor.)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

Insert proposed ITS 5.6.6 → M.9

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6-7-91

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

d. SEISMIC INSTRUMENTATION. Specifications 3.3.3.3 and 4.3.3.3.2.

e. METEOROLOGICAL INSTRUMENTATION. Specification 3.3.3.4.

f. Deleted.

g. LOOSE PARTS MONITORING SYSTEMS. Specification 3.3.3.9.

h. Deleted.

i. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

j. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

k. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

l. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

m. Deleted.

n. RADIOACTIVE EFFLUENTS. As required by the ODCM.

o. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

p. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

q. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

(A.33)

(A.1) ITS 5.0

ITS

~~r. Deleted~~ (A.33)

~~6.10 RECORD RETENTION~~ (A.1)

~~Section 6.10, "Record Retention," has been relocated to the Operational Quality Assurance Program.~~

A.1

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ITS 5.0

ITS

(A.1)

ADMINISTRATIVE CONTROLS at 30 centimeters from the Radiation Source or from any Surface Penetrated by the Radiation 2-17-94 (A.31)

6.11 RADIATION PROTECTION PROGRAM

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~ (A.8)

or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures (L.16) (M.4)

6.12 HIGH RADIATION AREA

5.7.1.a 6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less (A.31)
5.7.2.a than 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*. Any
5.7.1.b 5.7.2.b individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

5.7.1.d

5.7.1.d.1 a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

5.7.1.d.2 5.7.2.d.1 b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. ~~Entry into such~~
5.7.1.e 5.7.2.e INSERT 1 ~~areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.~~ (M.16) (M.19)

5.7.1.d.4(i) 5.7.2.d.5(i) INSERT 2 c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. ~~This individual shall be responsible for~~
5.7.1.d.4(ii) 5.7.2.d.3(ii) INSERT 3 ~~providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.~~ (L.28)

5.7.2 6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under
5.7.2.a.1 the administrative control of the ~~Shift Supervisor~~ on duty and/or the ~~Plant Health Physicist~~. (L.11)

radiation protection

5.7.2.a.2 except for 6.12.1.a

Doors and gates shall remain locked except during periods of personnel or equipment entry or exit. (A.17) (M.17)

radiation protection manager, or his or her designee

or continuously guarded (L.23)

at 30 centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (A.31)

only 5.7.1.c

or personnel continuously escorted by such individuals may (L.17)

5.7.1.c
5.7.2.c * ~~Health Physics~~ personnel ~~shall~~ be exempt from the RWP issuance requirement during the performance of their assigned ~~radiation protection~~ duties, provided they comply with approved radiation protection procedures for entry in high radiation areas. (L.11) (L.17)

radiation protection

INSERT proposed 5.7.1.d.3 and 5.7.2.d.2 (L.27)

INSERT proposed 5.7.2.d.4 (L.29)

INSERT proposed 5.7.2.f (L.13)

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Rev. 0

INSERT 1

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

INSERT 2

A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

INSERT 3

(For 5.7.1.d.4)

(i) ...that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

(For 5.7.2.d.3)

(i) ...that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

ITS

(A.1)

ITS 5.0

6.13 DELETED

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.r. This documentation shall contain:

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

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

2. Shall become effective after review and acceptance by the SNSOC and the approval of the Site Vice President.

(L.32)

UNIT 2

(A.1)

ITS 5.0

ITS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

plant manager

5.1.1 6.1.1 The ~~Site Vice President~~ shall be responsible for overall facility operation. ~~In his absence, the Manager - Station Operations and Maintenance shall be responsible for overall facility operation.~~ During ~~the~~ absence ~~of both~~, the ~~Site Vice President~~ shall delegate in writing the succession to this responsibility. his (L.4)

INSERT 1 (M.1)

5.1.2 6.1.2 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function and shall be the only individual that may direct the licensed activities of licensed operators. ~~A management directive to this effect, signed by the Senior Vice President - Nuclear, shall be reissued to all station personnel on an annual basis.~~ (L.5)

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 Onsite and Offsite Organization

5.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the UFSAR /QA Plan INSERT 2 (M.7) (L.6)

plant manager

5.2.1.b b. The ~~Site Vice President~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

a specified corporate officer

5.2.1.c c. ~~The Vice President - Nuclear Operations~~ 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. (L.6)

individuals

5.2.1.d d. The ~~management position~~ responsible for training of the operating staff and ~~the management position responsible~~ for the quality assurance functions shall have sufficient organizational freedom including sufficient independence from ~~cost and schedule when opposed to safety considerations.~~ (M.15)

may report to the appropriate onsite manager; however, these individuals

operating pressures

INSERT 1

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.

INSERT 2

including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications may report to the appropriate onsite manager, however, these individuals

A.1

Sufficient organizational freedom to ensure their independent from operating pressures

M.15

ITS

ADMINISTRATIVE CONTROLS

individuals

5.2.1.d e. The management position responsible for health physics shall have direct access to that onsite individual having responsibility for overall facility management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

FACILITY STAFF

include

5.2.2 6.2.2 The Facility organization shall be as shown in the UFSAR. L.3

5.2.2.a a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. ~~At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODES 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.~~ A.34

5.2.2.c c. A health physics technician# shall be onsite when fuel is in the reactor.

radiation protection

d. ~~ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~ A.9

5.2.2.c # The health physics technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions. L.11

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~~6.2.3 STATION NUCLEAR SAFETY (SNS)~~

~~FUNCTION~~

~~6.2.3.1 SNS shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources which may indicate areas for improving plant safety.~~ LA.4

~~COMPOSITION~~

~~6.2.3.2 SNS shall be composed of at least five dedicated, full-time engineers located onsite.~~

~~RESPONSIBILITIES~~

~~6.2.3.3 SNS shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.~~

~~6.2.3.4 SNS shall disseminate relevant operational experience.~~

~~AUTHORITY~~

~~6.2.3.5 SNS shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Manager - Station Safety and Licensing.~~

6.2.4 SHIFT TECHNICAL ADVISOR

An individual

Unit operations shift crew

5.2.2.f 6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit. A.28

in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the

This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. M.12

~~* Not responsible for sign-off function.~~ LA.4

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TABLE 6.2-1[a]

MINIMUM SHIFT CREW COMPOSITION

Total Staffing Requirements for Station Operation

With Either or Both Units in Mode 1, 2, 3 or 4

A.2

POSITION - NUMBER			CONDITIONS
SS	-	ONE	(Shift Supervisor may fulfill duties for both units).
SRO	-	ONE	(If ONE unit is in MODE 5, 6 OR DEFUELED, Senior Reactor Operator is assigned to the Unit in MODE 1, 2, 3 or 4). L.8
RO	-	THREE	(ONE Reactor Operator is assigned to each unit PLUS one is shared by both units).
AO	-	FOUR	(TWO Auxiliary Operators are assigned to each unit).
STA	-	ONE	(Shift Technical Advisor may fulfill duties for both units).

5.2.2.a

With Both Units in Mode 5 or 6 (or DEFUELED) L.9

A.2

POSITION - NUMBER			CONDITIONS
SS	-	ONE	(Shift Supervisor may fulfill duties for both units).
SRO	-	NONE	
RO	-	TWO	(ONE Reactor Operator is assigned to each unit). L.8
AO	-	TWO One	(ONE Auxiliary Operator is assigned to each unit). L.9
STA	-	ONE	(Shift Technical Advisor may fulfill duties for both units).

5.2.2.a

a - This Table and Table 6.2.1 of Unit 1 Technical Specifications represent Total Station Staffing and ARE NOT ADDITIVE. A.2

ITS

TABLE 6.2-1 (Continued)

A.12

SS	–	Shift Supervisor with a Senior Reactor Operators License on Unit 2.
SRO	–	Individual with a Senior Reactor Operators License on Unit 2.
RO	–	Individual with a Reactor Operators License on Unit 2.
AO	–	Auxiliary Operator
STA	–	Shift Technical Advisor

10CFR 50.54(m)(2)(ii) and 5.2.2.a and 5.2.2.f A.2

ITS 5.2.2.b Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 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 of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. L.1 A.2 L.1

ITS 5.1.2 During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command function. L.18 A.6 L.18

ITS 5.2.2.d ~~Procedures will be established to insure that NRC policy statement guidelines regarding working hours established for employees are followed. In addition, procedures will provide for documentation of authorized deviations from these guidelines and that the documentation is available for NRC review.~~ limit L.24 L.10

INSERT 5.2.2.d M.18

ITS 5.0

A.1

the individual providing advisory technical support to the unit operations shift crew A.28

5.3.1 6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1 (12/79 Draft)* for comparable positions, except for:

5.3.1 1. The Superintendent - Radiological Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.1 2. Incumbents in the positions of Shift Supervisor, Assistant Shift Supervisor (SRO), Control Room Operator - Nuclear (RO), and ~~Shift Technical Advisor~~ shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). ← INSERT A.29

5.2.2.e 3. The Superintendent Operations shall hold (or have previously held) a Senior Reactor Operator License for North Anna Power Station or a similar design Pressurized Water Reactor plant.

5.2.2.e 4. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station.

~~6.4 TRAINING~~

~~6.4.1 The Manager - Nuclear Training is responsible for ensuring that retraining and replacement training programs for the licensed facility staff meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). Also, a retraining and replacement training program for non-licensed facility staff shall meet or exceed the recommendations of Section 5 of ANS 3.1 (12/79 Draft)*.~~ L.19

~~6.5 REVIEW AND AUDIT~~

~~6.5.1 STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC) FUNCTION~~ LA.6

~~6.5.1.1 The SNSOC shall function to advise the Site Vice President on all matters related to nuclear safety.~~

~~* Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase."~~ L.19

INSERT

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

COMPOSITION

LA.6

6.5.1.2 The SNSOC shall be composed of:

Chairman: Manager - Station Safety and Licensing

Vice Chairman and Member: Manager - Station Operations and Maintenance

Member: Superintendent - Operations

Member: Superintendent - Maintenance

Member: Superintendent - Radiological Protection

Member: Superintendent - Engineering

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SNSOC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in SNSOC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SNSOC shall meet at least once per calendar month and as convened by the SNSOC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all new procedures required by Specifications 6.8.1 and 6.8.2, 2) all procedure changes that require a safety evaluation, 3) all programs required by Specification 6.8.4 and changes thereto, and 4) any other procedures or changes thereto as determined by the Site Vice President to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

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

d. Review of all proposed changes to Appendix "A" Technical Specifications and Appendix "B" Environmental Protection Plan. Recommended changes shall be submitted to the Site Vice President.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Operations and the MSRC.

f. Review of all REPORTABLE EVENTS and Special Reports.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee or Site Vice President.

i. Deleted.

j. Deleted.

LA.6

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k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Operations and the Management Safety Review Committee.

l. Review changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

m. Review of the Fire Protection Program and implementing procedures and shall submit recommended changes to the Site Vice President.

AUTHORITY

6.5.1.7 The SNSOC shall:

a. Provide written approval or disapproval of items considered under 6.5.1.6(a) through (c) above. SNSOC approval shall be certified in writing by either the Manager - Station Operations and Maintenance or the Manager - Station Safety and Licensing.

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

c. Provide written notification within 24 hours to the Vice President- Nuclear Operations and the Management Safety Review Committee (MSRC) of disagreement between the SNSOC and the Site Vice President; however, the Site Vice President shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

LA.6

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Site Vice President, Vice President-Nuclear Operations and the MSRC.

6.5.2 MANAGEMENT SAFETY REVIEW COMMITTEE (MSRC)

FUNCTION

6.5.2.1 The MSRC shall function to provide independent review of designated activities in the areas of:

a. Station Operations
b. Maintenance
c. Reactivity Management
d. Engineering
e. Chemistry and Radiochemistry
f. Radiological Safety
g. Quality Assurance Practices
h. Emergency Preparedness

COMPOSITION

6.5.2.2 The MSRC shall be composed of the MSRC Chairman and a minimum of four MSRC members. The Chairman and all members of the MSRC shall have qualifications that meet the requirements of Section 4.7 of ANSI/ANS 3.1-1979 Rev. 1 (Draft).

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the MSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in MSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants should be utilized as determined by the MSRC Chairman to provide expert advice to the MSRC.

LA.6

MEETING FREQUENCY

6.5.2.5 The MSRC shall meet at least once per calendar quarter.

QUORUM

6.5.2.6 The minimum quorum of the MSRC necessary for the performance of the MSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 50% of the MSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The MSRC shall be responsible for the review of:

a. Safety evaluations as programmatically discussed in the Updated Final Safety Analysis Report for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to assess the effectiveness of the safety evaluation program and to verify that the reviewed actions did not constitute an unreviewed safety question.

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

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

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

(A.1)

ITS 5.0

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

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

g. Events requiring written notification to the Commission.

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

i. A representative sample of reports and meetings minutes of the SNSOC.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the MSRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions. (LA.6)

b. The performance, training and qualifications of the entire facility staff.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50.

e. Any other area of facility operation considered appropriate by the MSRC or the Vice President - Nuclear Operations.

f. The Fire Protection Program and implementing procedures.

g. An independent fire protection and loss prevention inspection and audit shall be performed utilizing an outside qualified fire consultant.

h. The Radiological Environmental Monitoring Program and the results thereof.

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ITS

ADMINISTRATIVE CONTROLS

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes.

AUTHORITY

6.5.2.9 The MSRC shall report to and advise the Senior Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

LA.6

RECORDS

6.5.2.10 Records of MSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each MSRC meeting shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days of each meeting.

b. Reports of reviews with safety significant findings encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

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~~6.6 REPORTABLE EVENT ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~ A.19

~~a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

~~b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Vice President- Nuclear Operations and the MSRC.~~ LA.6

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President- Nuclear Operations and MSRC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President-Nuclear Operations and the MSRC within 14 days of the violation.

See ITS 2.0

6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

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

~~b. Refueling operations.~~ A.3

Insert proposed ITS 5.4.1.b → M.2

Insert proposed ITS 5.4.1.e → M.3

A.1

ITS 5.0

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.r. This documentation shall contain: L.32

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

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

b. Shall become effective after review and acceptance by the SNSOC and the approval of the Site Vice President.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

5.5.1 (2nd) a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.r. This documentation shall contain: A.11

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

5.5.1.a.2 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b (2nd) b. Shall become effective after review and acceptance by the SNSOC and the approval of the Site Vice President. plant manager LA.6 L.6

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a 1.17 The OFFSITE DOSE CALCULATION MANUAL (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/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also
5.5.1.b contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring activities Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.8 and 6.9.1.9. A.20

5.6.2 5.6.3

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1. See ITS 1.0

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

~~PROCESS CONTROL PROGRAM~~

~~1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.~~ L.32

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. See ITS 1.0

(A.1) ITS 5.0

c. Surveillance and test activities of safety related equipment. (A.3)

d. Security Plan implementation.

e. Emergency Plan implementation. (A.4)

5.4.1.d f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation. (L.32)

5.5.1 h. OFFSITE DOSE CALCULATION MANUAL implementation.

5.4.1.c i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975. (LA.1) (L.30)

6.8.2 Each new procedure of 6.8.1 above, except 6.8.1.d, 6.8.1.e, and 6.8.1.f, shall be reviewed and approved by the SNSOC prior to implementation as set forth in administrative procedures. (LA.6) Procedures of 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved as set forth in the facility's Security Plan, Emergency Plan, and section 6.5.1.6.m of the Technical Specifications, respectively. (L.30)

6.8.3 Procedure changes that require a safety evaluation shall also be reviewed and approved by SNSOC. All other changes shall be independently reviewed and approved as programmatically discussed in the Updated Final Safety Analysis Report. (LA.6) (L.30)

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

5.5.2 a. Primary Coolant Sources Outside Containment

A program ~~to reduce~~ 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 as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following: (L.12)

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. ~~In-Plant Radiation Monitoring~~ (LA.3)

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

5.5.9 c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off control point chemistry conditions, and (A.35)

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

5.5.3 d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

5.5.4 e. Radioactive Effluent Controls Program

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

5.5.4.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.4.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times 10 CFR ~~Part 20~~ 20.1001 - 20.2402, Appendix B, Table 2, Column 2. A.30

5.5.4.c 3) 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,

5.5.4.d 4) 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 Appendix I to 10 CFR Part 50,

5.5.4.e 5) 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. L.31

5.5.4.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.4.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

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

b) For Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr. to any organ.

Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

6.8.4.e Radioactive Effluent Controls Program (Cont.)

5.5.4.h 8) 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 Appendix I to 10 CFR Part 50,

5.5.4.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR 50,

5.5.4.j 10) 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 Part 190. L.25

INSERT →

~~f. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radio nuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~ LA.10

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

~~g. Configuration Risk Management Program~~

~~The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:~~ LA.8

~~1) Provisions for the control and implementation of a Level 1, at power, internal events, PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.~~

~~2) Provisions for performing an assessment prior to entering the LCO Action Statement for planned activities.~~

INSERT

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

Configuration Risk Management Program (continued)

3) Provisions for performing an assessment after entering the LCO Action Statement for unplanned entry into the LCO Action Statement.

4) Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action Statement.

5) Provisions for considering other applicable risk significant contributors such as Level 2 issue and external events, qualitatively or quantitatively.

Current risk-informed action statements include: Action 3.8.1.1.b; 3.4.3.2.A.2: 3.3.1.1: 3.3.2.1

LA.8

6.9 REPORTING REQUIREMENTS

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

ROUTINE REPORTS

5.3

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

A.24

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

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DESIGN FEATURES

5.5.5 5.7 COMPONENT CYCLIC or TRANSIENT LIMIT

5.7.1 The components identified in ~~Table 5.7-1~~ the UFSAR, Section 5.2, are designed and shall be maintained within the cyclic or transient design limits ~~of Table 5.7-1~~. LA.2

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at 100°F/hr and 200 cooldown cycles at 100°F/hr	Heatup cycle $-T_{avg}$ from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle $-T_{avg}$ from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at 200°F/hr	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER. (Full Power Trip)
	10 inadvertent pressurizer auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.

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REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3/4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

See CTS 3.4.10.1

SURVEILLANCE REQUIREMENTS

5.5.6 4.4.10.1.1 ~~In addition to the requirements of Specification 4.0.5,~~ the Reactor Coolant pump flywheels shall be inspected once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels. (A.22)

5.5.7 ~~4.4.10.1.2 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.~~ (LA.11)

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APPLICABILITY

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SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

See ITS 3.0

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

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

5.5.7 4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

(A.21)

5.5.7.a a. ~~Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice~~ testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

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APPLICABILITY

5.5.7 SURVEILLANCE REQUIREMENTS (Continued)

5.5.7a b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: A.21

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and 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

A.13

5.5.7.b c. The provisions of Specification 4.0.2 SR 3.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities. A.21 A.20

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.32

5.5.7.d e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

Insert proposed ITS 5.5.7.c L.20

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5.5.9 Steam Generator (SG) Tube Surveillance Program

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REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

~~3.4.5 Each steam generator in a non-isolated reactor coolant loop shall be OPERABLE.~~

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

~~ACTION:~~

~~With one or more steam generators in non-isolated reactor coolant loops inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

See ITS 3.4.13

~~SURVEILLANCE REQUIREMENTS~~

The provisions of SR3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies

~~4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the required Specification 4.0.5.~~

A.7

5.5.8.1 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table ~~4.4-1~~ 5.5.8-1.

5.5.8.2 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.4-2~~ 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification ~~4.4.5.3~~ 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification ~~4.4.5.4~~ 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

5.5.8.2.b b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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5.5.8.2.b.1 1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%, and

5.5.8.2.b.2 2. Tubes in those areas where experience has indicated potential problems.

5.5.8.4 A.20

5.5.8.2.b.3 3. A tube inspection (pursuant to Specification ~~4.4.5.4~~a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.5.8.2.c c. The tubes selected as the second and third samples (if required by Table ~~4.4-2~~) during each inservice inspection may be subjected to a partial tube inspection provided: A.20

5.5.8-2

5.5.8.2.c.1 1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

5.5.8.2.c.2 2. The inspections include those portions of the tubes where imperfections were previously found.

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

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

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

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5.5.8.3 4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

5.5.8.3.a a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

5.5.8-2

5.5.8.3.b b. If the results of the inservice inspection of a steam generator conducted in accordance with Table ~~4.4-2~~ at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification ~~4.4.5.3.a~~; the interval may then be extended to a maximum of once per 40 months. 5.5.8.3.a

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5.5.8.3.c c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.4-2~~ during the shutdown subsequent to any of the following conditions: 5.5.8-2

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification ~~3.4.6.2~~. 3.4.14

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A major steam line or feedwater line break.

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REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.4 4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

3. Degraded Tube means a tube containing imperfections greater than 20% of the nominal wall thickness caused by degradation.

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

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

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

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in ~~4.4.5.3.c~~ 5.5.8.3.c, above. (A.20)

8. Tube Inspection means an inspection of the steam generator tube from the point of entry on ~~the~~ hot leg side, completely around the U-bend to the top support of the cold leg side.

5.5.8.4

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table $\boxed{4.4-2}$. 5.5.8-2 A.20

5.6.7

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of an imperfection.
3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

NORTH ANNA - UNIT 2

TABLE ~~4.4-1~~ 5.5.8-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that 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 sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

Table 5.5.8-2

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. ~~Prompt notification to NRC pursuant to specification 6.9.1~~	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. ~~Prompt notification to NRC pursuant to specification 6.9.1~~	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

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ITS INSERT

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE: See ITS 3.7.10

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by: LA.5

5.5.10.a
5.5.10.b
1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e and f.). LA.5

5.5.10.c
2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.10.a
5.5.10.b
3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

5.5.10.c
c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%. LA.5

d. At least once per 18 months by: LA.5

5.5.10.e
1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber assembly is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

INSERT

Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at frequencies in general conformance with, and in accordance with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of, Regulatory Guide 1.52, Revision 2, and ANSI N510-1975.

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

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

5.5.10.a e. After each ~~complete~~ or partial ~~replacement~~ of a HEPA ~~filter~~ bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.5

5.5.10.b f. After each ~~complete~~ or partial ~~replacement~~ of a charcoal ~~adsorber~~ bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.5

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 1) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

See ITS 3.7.11

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PLANT SYSTEMS

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two safeguards area ventilation systems (SAVS) shall be OPERABLE with:

a. One SAVS exhaust fan, and

b. One auxiliary building HEPA filter and charcoal adsorber assembly (shared with Unit 1).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one SAVS inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SAVS system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Initiating, from the control room, flow through the auxiliary building HEPA filter and charcoal adsorber assembly and verifying that the SAVS operates for at least 10 hours with the heater on.

b. ~~At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:~~ LA.5

1. Verifying that the ECCS cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6,300 ~~cfm~~ ± 10% ~~(except as shown in Specifications 4.7.8.1e. and f.)~~ M.21

5.5.10.a&b

LA.5

nominal accident flow for a single train actuation M.21

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Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber.

(M.21)

SURVEILLANCE REQUIREMENTS (cont'd)

ITS

5.5.10.c 2. Verify~~ing, within 31 days after removal,~~ that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70 ~~95~~%. (LA.5) (L.33)

5.5.10.a 5.5.10.b 3. Verifying ~~a system~~ of one ECCS PREACS train provides greater than the minimum required cooling flow for ECCS equipment flow rate ~~of 6,300 cfm ± 10% during operation~~ when tested in accordance with ANSI N510-1975. (M.21)

5.5.10.c c. ~~Within 31 days of completing 720 hours of charcoal adsorber operation,~~ verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~%. 70 (LA.5) (L.33)

d. ~~At least once per 18 months by:~~ (LA.5)

5.5.10.d 1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is less than ~~6~~ 5 inches Water Gauge while operating the ~~ventilation system~~ ECCS PREACS at a flow rate of ~~6,300 cfm ± 10%~~ ≤ 39,200 cfm (M.21)

2. Verifying that on a Containment Pressure–High-High Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly. (See ITS 3.7.12)

5.5.10.a e. ~~After each complete or partial replacement of a HEPA filter bank by~~ verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. (LA.5) (M.21)

5.5.10.b f. ~~After each complete or partial replacement of a charcoal adsorber bank by~~ verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~ One ECCS PREACS train at nominal accident flow. (LA.5) (M.21)

A.1

~~Specifications 3/4.11.1.1 through 3/4.11.1.3 have been deleted~~

ITS 5.0

7-19-90

A.1

~~Specifications 3/4.11.2.1 through 3/4.11.2.4 have been deleted~~

A.1

ITS 5.0

ITS

INSERT →

9-25-91

M.11

RADIOACTIVE STORAGE

3/4.11.2 GAS STORAGE

EXPLOSIVE GAS MIXTURE

LIMITING CONDITIONS FOR OPERATION

5.5.11.a 3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration could exceed 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the affected waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the affected waste gas decay tank greater than 4% volume immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 4% by volume without delay, then continue with Action "a" above.

c. With the requirements of Action "a" not satisfied, immediately suspend all additions of waste gases to the affected tank until the oxygen concentration is restored to less than 2% by volume, and submit a Special Report to the commission pursuant to Specification 6.9.2 within the next 30 days outlining the following:

    1. The cause of the waste gas decay tank exceeding the 2% oxygen limit,

    2. the reason why the oxygen concentration could not be returned to within limits, and

    3. actions taken and the time required to return the oxygen concentration to within limits.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

LA.7

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be determined to be within the above limits by continuously monitoring the waste gases in the inservice waste gas decay tank with the oxygen monitor required OPERABLE by Table 3.3-14 of Specification 3.3.3.11.

Insert Proposed ITS 5.5.11.a

M.11

INSERT

Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Decay Tanks, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The Liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Release due to Tank Failure."

RADIOACTIVE STORAGE

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

5.5.11.b

~~3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ≤ 25,000 curies noble gases (considered as Xe-133).~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

LA.7

~~SURVEILLANCE REQUIREMENTS~~

~~4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is ≤ 1.0 µCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 µCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.~~

Insert proposed ITS 5.5.11.b

M.11

A.1

ITS 5.0

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ITS

RADIOACTIVE STORAGE

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

INSERT 5.5.11.c

M.11

5.5.11.c 3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. LA.7

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank**

LA.7

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

LA.7

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per week when radioactive materials are being added to the tank.

*This is a shared system with Unit 1.

**Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste ion exchanger system.

INSERT →

A.18

INSERT

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.

A.1

Specifications 3/4.11.3 through 3/4.11.4 have been deleted

Insert proposed ITS 5.5.12 M.5

Insert proposed ITS 5.5.13 M.6

Insert proposed ITS 5.5.14 L.14

Insert proposed ITS 5.5.15 A.10

A.1

ITS 5.0

ITS

02-09-96

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4

See ITS 3.6.1

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves, secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1. See ITS 3.6.3

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3. See ITS 3.6.2

5.5.15 c. ~~After each closing of the equipment hatch, by leak rate testing the equipment hatch seals, with gas at P_a, greater than or equal to 44.1 psig. Results shall be evaluated against the criteria of Specification 3.6.1.2.b as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995.~~ A.2b

d. ~~Each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line.~~

* Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked sealed or otherwise sealed in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such surveillance need not be performed more often than once per 92 days. See ITS 3.6.3

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

5.5.15 3.6.1.2 Containment leakage rates shall be limited to:

5.5.15.c 5.5.15.d.1 a. An overall integrated leakage rate of less than or equal to L_a, 0.1 percent by weight of the containment air per 24 hours, at the calculated peak containment pressure P_a,

5.5.15.b greater than or equal to 44.1 psig. The containment design pressure is 45psig. M.20

5.5.15.d.1 b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, greater than or equal

5.5.15.b to 44.1 psig.

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

ACTION:

5.5.15.d.1 With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F. <See ITS 3.6.1>

SURVEILLANCE REQUIREMENTS

5.5.15.a 4.6.1.2 The containment and containment penetrations shall be tested by performing leakage rate testing as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. The provisio~~ns~~ of Specification 4.0.2 are ~~not~~ applicable. A.16

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

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program. A.36

A.1

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ITS 5.0

ITS

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

5.5.15.d.2.a b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a greater than or equal to 44.1 psig.

See ITS 3.6.2

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.+
2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.
3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
4. The provisions of Specification 3.0.4 are not applicable.

b. With a containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.2

SURVEILLANCE REQUIREMENTS

5.5.15.a 4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By performing leakage rate testing as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. The provisions of Specification 4.0.2 are not applicable.

5.5.15.f

b. At least once per refueling outage by verifying that only one door in each air lock can be opened at a time.

See ITS 3.6.2

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

A.16

+ Entry to repair the inner air lock door, if inoperable, is allowed.

ITS

ADMINISTRATIVE CONTROLS

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed. (L.7)

ANNUAL REPORTS[1/] for the Steam Generator Tube Inspection Report and by Apr. 30 of each year for the Occupational Radiation Exposure Report (L.15)

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. ~~The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ (A.25)

6.9.1.5 Reports required on an annual basis shall include: collective deep dose equivalent (reported in person-rem)

5.6.1 a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed receiving ~~exposures~~ an annual deep dose equivalent greater than 100 mrem/yr and their associated ~~man-rem exposure~~ according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, electronic dosimeter or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total ~~whole body dose~~ deep dose equivalent received from external sources should be assigned to specific major work functions. (M.14) (L.21) (M.14)

Note: 5.6.1, 5.6.2, + 5.6.3 1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

5.6.1 2/ This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20.

ITS

5.0 ADMINISTRATIVE CONTROLS (Continued)

5.6.2 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

INSERT →

M.8

Note 5.6.2 * A single submittal may be made for a multiple unit station.

INSERT

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 ANNUAL RADIOLOGICAL EFFLUENT RELEASE REPORT*

6.9.1.9 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. 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 (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

shall M.10

Note 5.6.3 * A single submittal may be made for a multiple unit station. The submittal ~~should~~ shall combine those sections that are 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.

5.6.7 b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b). 5.6.7 A.20

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit. L.2

MONTHLY OPERATING REPORT

5.6.4 6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th day of each month following the calendar month covered by the report. L.26 A.14

5.6.5 CORE OPERATING LIMITS REPORT

5.6.5.a 6.9.1.7.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Safety Limits,
2. Shutdown Margin,

1. Moderator Temperature Coefficient ~~BOC and EOC limits, and 300 ppm and 60 ppm surveillance limits for Specification 3/4.1.1.4,~~

2. Shutdown Bank Insertion Limit ~~for Specification 3/4.1.3.5,~~

3. Control Bank Insertion Limits ~~for Specification 3/4.1.3.6,~~

4. Axial Flux Difference limits ~~for Specification 3/4.2.1,~~

5. Heat Flux Hot Channel Factor, ~~K(Z), N(Z) for Specification 3/4.2.2,~~ and

6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier, ~~for Specification 3/4.2.3.~~

9. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
10. RCS Pressure, Temperature, and Flow DNB limits, and
11. Boron Concentration.

A.37

5.6.5.b 6.9.1.7.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC ~~as identified in 6.9.1.7.e.~~ described in the following documents.

5.6.5.c 6.9.1.7.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d 6.9.1.7.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A.14

6.9.1.7.e REFERENCES

5.6.5.b

1. VEP-FRD-42, ~~Rev. 1-A,~~ "Reload Nuclear Design Methodology," ~~September 1986.~~ LA.9

~~(Methodology for LCO 3.1.1.4 - Moderator Temperature Coefficient, LCO 3.1.3.5 - Shutdown Bank Insertion Limit, LCO 3.1.3.6 - Control Bank Insertion Limits, LCO 3.2.2 - Heat Flux Hot Channel Factor, LCO 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~ LA.9

A.1

ITS 5.0

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ITS

5.6.5.b

2a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

LA.9

2b. 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 IN W ECCS EVALUATION MODEL", JULY, 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2f. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY REPORT," -REFERENCE CORE June 1990 (W Proprietary).

A.27

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor.)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

M.9

INSERT ITS 5.6.6

A.1

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. Deleted.

d. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

e. Deleted.

f. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

g. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

h. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

i. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

j. Deleted.

k. Deleted.

l. RADIOACTIVE EFFLUENTS. As required by the ODCM.

m. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

n. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

o. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

p. Deleted.

(A.33)

A.1

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6.10 RECORD RETENTION

Section 6.10, "Record Retention," has been relocated to the Operational Quality Assurance Program.

A.1

A.1

ITS 5.0

A.31

at 30 centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure. A.8

or equivalent L.16

That includes specification of radiation dose rates in the immediate work areas) and other appropriate radiation protection equipment and measures M.4

6.12 HIGH RADIATION AREA

5.7.1.a
5.7.2.a
5.7.1.b
5.7.2.b
5.7.1.d

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 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.* 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.31

5.7.1.d.1 a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

5.7.1.d.2
5.7.2.d.1
5.7.1.e
5.7.2.e

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them. M.16

INSERT 1

INSERT 2 M.19

5.7.1.d.4(i)
5.7.2.d.3(i)
5.7.1.d.4(ii)
5.7.2.d.3(ii)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit. L.28

INSERT 3

except for 6.12.1.a M.17

5.7.2 6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist. A.31

5.7.2.a.1 L.11

radiation protection

at 30 centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

radiation protection manager, or his or her designee

or continuously guarded L.23

Doors and gates shall remain locked except during periods of personnel or equipment entry or exit A.17

5.7.2.a.2
5.7.1.c
5.7.2.c

radiation protection

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry in high radiation areas. L.11 L.17

Only 5.7.1.c M.13

Insert proposed 5.7.1.d.3 and 5.7.2.d.2 L.27

Insert proposed 5.7.2.d.4 L.29

page 66 of 69

Rev 0

Insert proposed 5.7.2.f L.13

INSERT 1

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

INSERT 2

A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

INSERT 3

(For 5.7.1.d.4)

(i) ...that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

(For 5.7.2.d.3)

(i) ...that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

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ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Table 6.2-1 states Shift Supervisor (SS), Senior Reactor Operator (SRO) and Reactor Operator (RO) manning requirements. The ITS does not include these manning requirements. This changes the CTS by not including manning requirements already required by 10 CFR 50.54(m)(2)(i).

The purpose of CTS Table 6.2-1 is to specify the minimum shift crew composition consistent with 10 CFR (m)(2)(i). This change is acceptable because 10 CFR 50.54 (m)(2)(ii) already states this required composition. This change is designated administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.8.1.b requires written procedures be established, implemented and maintained covering refueling operations. CTS 6.8.1.c requires written procedures be established, implemented and maintained covering surveillance and test activities of safety related equipment. ITS 5.4.1.a requires written procedures shall be established, implemented and maintained to cover the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. This changes the CTS by deleting the specific wording of 6.8.1.b and 6.8.1.c, which is already addressed by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 and is committed to in CTS 6.8.1.a and ITS 5.4.1.a.

This change is acceptable because the recommendations of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 already require procedures for refueling operations and for surveillance tests for safety related activities. This change is designated administrative because it does not result in technical changes to the CTS.

A.4 CTS 6.8.1.d and CTS 6.8.1.e require written procedures be established, implemented, and maintained to address implementation of the Security Plan and the Emergency Plan. The ITS does not contain these requirements. This changes the CTS by deleting the specific reference to the Security Plan and the Emergency Plan because they are already required by 10 CFR 50.54(p) and 10 CFR 50.54(q), respectively.

This change is acceptable because the requirements for implementation of the Security and Emergency Plans are contained in 10 CFR 50.54(p) and 10 CFR 50.54(q). This change is designated administrative because it does not result in technical changes to the CTS.

A.5 ITS 5.5.10, Ventilation Filter Testing Program (VFTP), states, "A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at frequencies in general conformance with, and in accordance with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of, Regulatory Guide 1.52, Revision 2, and ANSI N510-1975." CTS 4.7.7.1 (Control Room Emergency Ventilation System) and 4.7.8.1 (Safeguards Area Ventilation System) include requirements for ventilation filter testing in accordance with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975. This changes the CTS by consolidating existing ventilation requirements in a single program.

The purpose of CTS 4.7.7.1 and 4.7.8.1 is to specify the Surveillance Requirements for the ventilation filter testing in accordance with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975. This change is acceptable because it retains existing ventilation testing requirements in a single program in the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.6 CTS Table 6.2-1 states, "During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, and individual with a valid RO license...shall be designated to assume the Control Room command function." ITS 5.1.2 adds the option for a person with an active SRO license to assume the Control Room command function. This changes the CTS by clarifying that an SRO may also assume the Control Room command function.

This change is acceptable because a person with an SRO license is always allowed to assume the Control Room command function. The CTS and ITS allowance to use an RO is an exception to that requirement. This change is designated administrative because it does not result in technical changes to the CTS.

A.7 ITS 5.5.8 states, "The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies." CTS 3.4.5 does not include such a reference because CTS 4.0.2 already applies to CTS 3.4.5. This changes the CTS by adding an explicit reference to the ITS for an allowance provided without the reference in the CTS.

This change is acceptable because the added phrase retains an existing allowance, and is only required because of the change in format from CTS to ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.8 CTS 6.11, Radiation Protection Program, states, "Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure." The ITS does not include a requirement for a Radiation Protection Program. This changes the CTS by removing references to requirements already required by 10 CFR Part 20.

This change is acceptable because the requirements of 10 CFR Part 20 are already required to be met. This change is designated administrative because it does not result in technical changes to the CTS.

A.9 CTS 6.2.2.d states, "ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation." ITS 5.2.2 does not contain this requirement. 10 CFR 50.54(m)(2)(iv) states, "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person." This changes the CTS 6.2.2.d by deleting this information because it is already a requirement in accordance with 10 CFR 50.54 (m)(2)(iv).

The purpose of CTS 6.2.2.d is to ensure the presence of a licensed SRO or licensed SRO limited for fuel handling who has no other concurrent responsibilities during this operation. This change is acceptable because it is a duplication of 10 CFR 50.54 (m)(2)(iv), and the requirement is retained, but not in the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.10 CTS 4.6.1.1, CTS 4.6.1.2, CTS 3.6.1.3, and CTS 4.6.1.3 specify the leakage rate requirements for Containment Integrity and the Containment Air Locks. ITS 5.5.15, Containment Leakage Rate Testing Program, specifies the leakage rate requirements for the Containment and Containment Air Locks within the Containment Leakage Rate Testing Program. This changes the CTS by moving the leakage rate acceptance criteria for Containment Integrity and Containment Air Locks in the CTS to ITS 5.5.15, "Containment Leakage Rate Testing Program."

This change is acceptable because the same containment leakage rate requirements are being applied, but as a program in ITS 5.5.15 instead of individual LCOs and Surveillance Requirements. This change is designated administrative because it does not result in technical changes to the CTS.

A.11 CTS 6.15 states, "Changes to the ODCM: a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.r." ITS 5.5.1 states, "Licensee initiated changes to the ODCM: a. Shall be documented and records of reviews performed shall be retained." This changes the CTS by not including a reference to how the records are to be retained.

This change is acceptable because referenced requirement CTS 6.10.2.r was removed from the CTS by North Anna amendment 208 (Unit 1) / 189 (Unit 2). This change is designated administrative because it does not result in technical changes to the CTS.

A.12 CTS Table 6.2-1 lists acronym definitions for shift manning. These acronyms are defined as appropriate in parts of ITS 5.0, and the ITS does not include a consolidated list. This changes the CTS by deleting the consolidated acronym list and defining them as needed in ITS 5.0.

This change is acceptable because the acronyms are adequately defined where appropriate in ITS 5.0, and it is not necessary to have a consolidated list. This change is designated administrative because it does not result in technical changes to the CTS.

A.13 CTS 4.0.5.b does not specify a biennial or every 2 years frequency of "at least once per 731 days." ITS 5.5.7 includes a biennial or every 2 years frequency of "at least once per 731 days." This changes the CTS 4.0.5 by incorporating the ASME Boiler and Pressure Vessel Code biennial or every 2 years frequency of "at least once per 731 days."

The purpose of CTS 4.0.5.b is to specify the required frequencies for performing inservice testing activities associated with ASME Boiler and Pressure Vessel Code. This change is acceptable because it adds the ASME Boiler and Pressure Vessel Code biennial or every 2 years frequency of "biennially or every 2 years" without adding any new requirements. This change is designated administrative because it does not result in technical changes to the CTS.

A.14 CTS 6.9.1.7.d requires the COLR to be provided to the, "NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector." CTS 6.9.1.6 requires the Monthly Operating Report be submitted to, "the Director of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement." ITS 5.6.5.d requires the COLR be provided to the NRC. ITS 5.6.4 requires the Monthly Operating Report be submitted. This changes the CTS by removing the specifics regarding distribution of the reports to the NRC, which is addressed by 10 CFR 50.4.

This change is acceptable because the distribution of written communications to the NRC is governed by 10 CFR 50.4, and duplication in the Technical Specifications is unnecessary. This change is designated administrative because it does not result in technical changes to the CTS.

A.15 Unit 2 CTS Table 4.19-2, Steam Generator Tube Inspection, 1st Sample Inspection, C-3 result, and 2nd Sample Inspection, Additional SG is C-3, Action Required includes, "Special Report." ITS Table 5.5.8-2 does not include a statement requiring prompt NRC notification. ITS 5.6.7.c states, "Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence." This changes the CTS by removing a reporting reference that is required by other sections of the Technical Specifications.

This change is acceptable because a duplicate reporting requirement is deleted that is addressed by other Technical Specifications. This change is designated administrative because it does not result in technical changes to the CTS.

A.16 CTS 4.6.1.2 and CTS 4.6.1.3 regarding the containment and containment penetrations, and each containment air lock, respectively, state they shall, "...be tested by performing leakage rate testing as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. The provisions of Specification 4.0.2 are not applicable." ITS 5.5.15, Containment Leakage Rate Testing Program, does not include the statement that the provisions of Specification 4.0.2 are not applicable, but states, "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J." This changes the CTS by removing a statement that part of Section 3.0 does not apply to this testing requirement which is being moved to Section 5.0 because Section 3.0 is understood to not apply to Section 5.0.

The purpose of the CTS 4.6.1.2 and CTS 4.6.1.3 statements that the provisions of Specification 4.0.2 are not applicable is to require the testing frequencies for containment and containment penetrations to remain as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. The NRC and industry position is that Section 3.0 does not apply to Section 5.0. The statement, "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J," was added to avoid any possible confusion. Therefore, the requirements of CTS 4.0.2 continue to not be applicable to the containment and containment penetration leakage testing requirements, but the format is changed to accommodate moving the testing requirements to Section 5.0. This change is designated administrative because it does not result in technical changes to the CTS.

A.17 ITS 5.7.2.a.2 states, in reference to entryways to high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any Surface Penetrated by the Radiation, "Doors and gates shall remain locked except during periods of personnel or equipment entry or exit." The CTS does not include such a statement. This changes the CTS by adding a clarification that the door and gate barriers may be opened for entry and exit.

This change is acceptable because it clarifies that entry and exit through these barriers is allowed under specified controls, as is the case in the CTS. This change is designated administrative because it does not result in technical changes to the CTS.

A.18 ITS 5.5.11 states, "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." CTS 3.11.1 and CTS 3.11.2 did not include such requirements because CTS 4.0.2 and 4.0.3, which are equal to ITS SR 3.0.2 and SR 3.0.3, already apply to CTS 3.11.1 and CTS 3.11.2. This changes the CTS by adding a reference for an allowance because it must be stated that the existing allowance applies for testing in Section 5.0.

This change is acceptable because the added phrase retains existing allowances, and is only required because of the change in format from the CTS to the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.19 CTS 6.6.1 states, "The following actions shall be taken for REPORTABLE EVENTS: A. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and..." ITS 5.0 does not include these requirements. This changes the CTS by deleting requirements already required by 10 CFR 50.73.

This change is acceptable because the reporting requirements of 10 CFR 50.73 are still required without a reference in the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.20 CTS 1.17, 4.0.5.c, 4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, 6.9.1.5.b, and 6.12.2 include references to other CTS requirements. The ITS modifies these to ITS references or appropriate requirements. This changes the CTS by making appropriate references in the ITS.

This change is acceptable because it makes appropriate reference changes for the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.21 CTS 4.0.5.a states, "Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). CTS 4.0.5 and CTS 4.0.5.c reference inservice inspection requirements for ASME Code Class 1, 2, and 3 components. ITS 5.5.7 does not include the statement in CTS 4.0.5.a and does not include references to inservice inspection. This changes the CTS by not including a reference to 10 CFR

50.55a requirements or references to ASME Code Class 1, 2, and 3 inservice inspection. The 10 CFR 50.55a requirements are still applicable without the reference, and inservice inspection is understood to be part of ASME Code Class 1, 2, and 3 inservice testing.

This change is acceptable because the inservice inspection requirements and 10 CFR Part 50 requirements are still applicable and referencing them separately is unnecessary. This change is designated administrative because it does not result in technical changes to the CTS.

A.22 CTS 4.4.10.1.1 states, "In addition to the requirements of Specification 4.0.5, the Reactor Coolant pump flywheels shall be inspected..." ITS 5.5.6 does not include the reference to Specification 4.0.5, which is ITS 5.5.7, Inservice Testing Program. This changes the CTS by not referencing CTS 4.0.5 requirements which are required regardless of the reference.

This change is acceptable because it deletes a reference to a requirement that has it's own criteria for application, regardless of the reference. This change is designated administrative because it does not result in technical changes to the CTS.

A.23 ITS 5.5.10 states, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies." CTS 4.7.7 and CTS 4.7.8 do not explicitly state these allowances, but they apply as CTS 4.0.2 and CTS 4.0.3, which are equal to ITS SR 3.0.2 and SR 3.0.3, because these allowances apply to all the CTS LCO Surveillance Requirements. This changes the CTS by explicitly invoking the allowances of ITS SR 3.0.2 and ITS SR 3.0.3 because the requirements have been moved to Section 5.0, and an explicit allowance is needed to retain the existing allowances.

This change is acceptable because it retains existing allowances by transferring them into ITS format. This change is designated administrative because it does not result in technical changes to the CTS.

A.24 CTS 6.9.1 states, "In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted." ITS 5.6 states, "The following reports shall be submitted in accordance with 10 CFR 50.4." This changes the CTS by referencing 10 CFR 50.4 as the reference for how to submit reports and excluding the remaining detail, which is already addressed in 10 CFR 50.4.

This change is acceptable because the reporting requirements are already established in 10 CFR 50.4, and do not need to be repeated in the ITS. This change is designated administrative because it does not result in technical changes to the CTS.

A.25 CTS 6.9.1.4 regarding annual reports states, "The initial report shall be submitted prior to March 1 of the year following initial criticality." The ITS does not include such a statement. This changes the CTS by deleting a requirement for report submissions that have already occurred and will not be repeated.

This change is acceptable because the one time report requirement has already been met and no longer needs to be specified. This change is designated administrative because it does not result in technical changes to the CTS.

A.26 CTS 4.6.1.1 states, "Primary CONTAINMENT INTEGRITY shall be demonstrated... c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals, with gas at P_a, greater than or equal to 44.1 psig. Results shall be evaluated against the criteria of Specification 3.6.1.2.b as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. d. Each time containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves in the containment purge lines and the containment vacuum ejector line." ITS 5.5.15, the Containment Leakage Rate Testing Program, states, "A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50, Appendix J, Option B, for Type A, B, and C testing, 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." ITS 5.5.15 does not include an explicit requirement for testing the equipment hatch, the containment purge lines, or the containment vacuum ejector line. This changes the CTS by deleting the explicit leak rate testing for the equipment hatch, the containment purge lines, and the containment vacuum ejector line because it is already required as part of ITS 5.5.15.

The purpose of CTS 4.6.1.1.c is to provide assurance that the equipment hatch is tested after each closing of the equipment hatch, prior to unit operation. The purpose of CTS 4.6.1.1.d is to provide assurance that the butterfly isolation valves in the containment purge lines each time containment integrity is established after vacuum has been broken. This change is acceptable because Regulatory Guide 1.163, dated September 1995, required by ITS 5.5.15, states that NEI 94-01, Revision 0, provides methods acceptable to the NRC for complying with 10 CFR Part 50, Appendix J, Option B. Section 10.2.1.3 of NEI 94-01 requires a Type B test be performed prior to the time containment integrity is required, if a containment penetration is opened. The equipment hatch and the butterfly isolation valves in the containment purge lines are containment penetrations, so they must be tested prior to the time containment integrity is required. This change is designated administrative because it does not result in technical changes to the CTS.

A.27 CTS 6.9.1.7.e.2f , References for the Core Operating Limits Report, states, "WCAP-12610, "VANTAGE+FUEL ASSEMBLY REPORT," June 1990 (W Proprietary)."

ITS 5.6.5.b.2f states, "VANTAGE+FUEL ASSEMBLY-REFERENCE CORE REPORT." This changes the CTS by correcting the reference to the title of WCAP-12610. Regarding deletion of, "June 1990 (W Proprietary)," see DOC LA.9.

This change is acceptable because it corrects the title of a reference used, without changing the reference. This change is designated administrative because it does not result in technical changes to the CTS.

A.28 CTS 6.2.4.1 states, "The Shift Technical Advisor shall serve in an advisory capacity to Shift Supervisor on matters..." CTS 6.3.1.2 states, "Incumbents in the positions of Shift Supervisor, Assistant Shift Supervisor (SRO), Control Room Operator – Nuclear (RO), and Shift Technical Advisor, shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4)." ITS 5.2.2.f states, "An individual shall provide advisory technical support to the unit operations shift crew..." ITS 5.3.1 states, "The SS, Assistant SS, Control Room Operator – Nuclear, and individual providing advisory technical support to the unit operations shift crew, shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4)." This changes the CTS by removing the Shift Technical Advisor title, and replacing the term Shift Supervisor with unit operations shift crew, though the requirement for the person with the specified responsibility remains the same.

This change is acceptable because the individual assigned to the responsibilities described still carries out the same tasks. The support provided is for the benefit of the unit operations shift crew, as well as the Shift Supervisor. This change clarifies that the assigned individual may provide the support directly to the Shift Supervisor or other members of the unit operations shift crew, but the result will be support for the crew as a whole in either case. This change is designated administrative because it does not result in technical changes to the CTS.

A.29 ITS 5.3.2 states, "For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m)." The CTS does not include such a statement. This changes the CTS by clarifying the relation between individuals referenced in 10 CFR 55.4, ITS 5.3.1, and 10 CFR 50.54(m).

This change is acceptable because it clarifies an existing relation between the Technical Specifications and regulations. This change is designated administrative because it does not result in technical changes to the CTS.

A.30 CTS 6.8.4.e.2 states that the program provided conforming with 10 CFR 50.36a includes, "Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times 10 CFR Part 20 Appendix B, Table 2, Column 2." ITS 5.5.4.b references 10 CFR 20.1001-20.2402.

This changes the CTS by referencing the specific portion of 10 CFR Part 20 that includes the referenced requirement.

This change is acceptable because it clarifies which regulatory requirement is referenced for meeting the Technical Specification requirement, but does not change the requirement. This change is designated administrative because it does not result in technical changes to the CTS.

A.31 CTS 6.12.1 applies for control of entry into high radiation areas in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr. CTS 6.12.2 applies for control of entry into high radiation areas in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. ITS 5.7.1 applies to controls for high radiation areas with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any Surface Penetrated by the Radiation. ITS 5.7.2 applies to controls for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any Surface Penetrated by the Radiation, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. This changes the CTS by deleting the reference to a high radiation area having radiation intensity in excess of 100 mrem/hr, and adds the criteria of, "at 30 centimeters from the radiation source or from any Surface Penetrated by the Radiation" to the parameter 1000 mrem/hr.

These changes are acceptable because the 100 mrem/hr definition for a high radiation area is already addressed by 10 CFR 20.1003, and the method of measuring the 1000 mrem/hr is clarified in terms of being measured from a point source and from a surface. This change is designated administrative because it does not result in technical changes to the CTS.

A.32 CTS 4.0.5.d states, "Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements." The ITS does not include an equivalent requirement. This changes the CTS by not explicitly stating that the inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

This change is acceptable because the inservice inspection and testing activities are still required by 10 CFR 50.55a, as appropriate, and ITS 5.5.7, the Inservice Testing Program. A specific reference to this fact is unnecessary. This change is designated administrative because it does not result in technical changes to the CTS.

A.33 CTS 6.9.2 requires special reports be submitted to the Regional Administrator, Region II, within time periods specified, and lists the CTS Specifications that require special reports to be submitted. The ITS does not require special reports to be prepared and submitted. This changes the CTS by deleting the references to the CTS Specifications requiring special reports be generated. Justification for disposition of each of the special report requirements is addressed by the ITS package for each respective CTS Specification.

The purpose of CTS 6.9.2 is to identify the specifications that require special reports to be submitted. This change is acceptable because the special reports are no longer required by the respective Specifications. Justification for disposition of each of the special report requirements is addressed by the ITS package for each respective CTS Specification. This change is designated administrative because it does not result in technical changes to the CTS.

A.34 CTS 6.2.2.b states, "At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODES 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room." The ITS does not include this phrase. This changes the CTS by deleting two requirements, both of which are addressed by 10 CFR 50.54.

10 CFR 50.54 (m)(2)(iii) states, "When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by a unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times." 10 CFR 50.54(k) states, "An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during operation of the facility." This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.54. This change is designated administrative because it does not result in technical changes to the CTS.

A.35 CTS 6.8.4.c(v) states that the secondary water chemistry monitoring program shall include, "Procedures defining corrective actions for all control point chemistry conditions." ITS 5.5.9.e states that the secondary water chemistry monitoring program shall include, "Procedures defining corrective actions for all off control point chemistry conditions." This changes the CTS by adding the word "off" to the term control point.

This change is acceptable because the intent of CTS 6.8.4(v) is to provide procedures for what to do when the control point chemistry conditions are not within limits, which is more accurately stated using the term "off control point." This change clarifies an existing requirement. This change is designated administrative because it does not result in technical changes to the CTS.

A.36 ITS 5.5.15.e states, "The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program." The CTS do not contain such a statement. This changes the CTS by stating that SR 3.0.3 applies because in the CTS the allowance in CTS 4.0.2, which is the same as ITS SR 3.0.3, already applies.

This change is acceptable because it retains the allowance in CTS 4.0.2, which must be explicitly stated for it to apply to a requirement in ITS Section 5.0. This change is designated administrative because it does not result in technical changes to the CTS.

A.37 CTS 6.9.1.7.a contains a list of the core operating limits established and documented in the Core Operating Limits Report (COLR). ITS 5.6.5.a includes additional core operating limits established and documented in the COLR. These are: Safety Limits, Shutdown Margin, Reactor Trip System Instrumentation – OTΔT and OPΔT Trip Parameters, RCS Pressure, Temperature, and Flow DNB Limits, and Boron Concentration. These limits had previously been addressed in other parts of the CTS, but are being moved to the COLR, and because of this are listed in ITS 5.6.5.a. The change also deletes references associating the core operating limits listed with other sections in the CTS. This changes CTS by adding core operating limits established and documented in the COLR because they are being moved there as part of changes to other parts of the CTS. Technical aspects of the changes are addressed by Discussions of Change for the respective individual specifications.

This change is acceptable because it administratively documents changes made to other parts of the CTS and the COLR. This change is designated administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 ITS 5.1.1 states, "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." The CTS does not include such a statement. This changes the CTS by adding a required action for the plant manager or his designee.

The purpose of the ITS 5.1.1 statement is to provide additional assurance that the plant manager has direct responsibility for overall unit operation. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with the ITS 5.2.1.b requirement, "The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant." This change is designated more restrictive because an additional requirement is added to the Technical Specifications.

M.2 ITS 5.4.1 states, "Written procedures shall be established, implemented, and maintained covering the following activities:...b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33." The CTS does not include this requirement. This changes the CTS by adopting a new requirement for emergency operating procedures.

The purpose of ITS 5.4.1.b is to ensure that written procedures are established, implemented, and maintained covering the emergency operating procedures to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This change is acceptable because it is consistent with an existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This change is designated more restrictive because it imposes a new requirement for procedures within the Technical Specifications.

M.3 ITS 5.4.1 states, "Written procedures shall be established, implemented, and maintained covering the following activities:...e. All programs specified in Specification 5.5." The CTS does not include this requirement. This changes the CTS by adopting a new requirement for procedures to address programs described in ITS 5.5.

The purpose of ITS 5.4.1.e is to ensure that written procedures are established, implemented, and maintained covering all programs specified in ITS 5.5. This change is considered acceptable because it requires procedures to address programs required by ITS 5.5. Some of the programs already have procedures, some already have procedures for parts of the programs and need a document to tie them together, and others will need a new procedure altogether. This change is designated more restrictive because it imposes new requirements for procedures within the Technical Specifications.

M.4 CTS 6.12.1 states, "...entrance [into a high radiation area] thereto shall be controlled by requiring issuance of a Radiation Work Permit." ITS 5.7.1.b and 5.7.2.b state, "Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures." This changes the CTS by specifying certain information is required to be in the RWP or equivalent. The addition of the option to use a means equivalent to the RWP is addressed in DOC L.16.

The purpose of the RWP requirement in CTS 6.12.1 is to ensure personnel entering a high radiation area have the information necessary to work safely in those areas from a radiation standpoint. This change is acceptable because it states specific information to be included in the RWP to accomplish the same goal, and requiring issuance of the RWP with the required information makes the information available. These changes are designated as more restrictive because additional information to be included in the RWP is required.

M.5 The CTS does not contain a diesel fuel oil testing program that controls the requirements for testing and maintaining the properties of both new and stored fuel oil. ITS 5.5.12 establishes a diesel fuel oil testing program to implement required testing of both new and stored fuel oil. This changes the CTS requirements by adding the requirement for a diesel fuel oil testing program.

The purpose of ITS 5.5.12 is to establish a diesel fuel oil testing program that sets specific limits and testing requirements on diesel fuel oil. This change is acceptable because the program includes sampling and testing requirements, and acceptance criteria, in accordance with applicable ASTM Standards that supports EDG OPERABILITY. The proposed sampling and testing requirements and acceptance criteria provide limits that, if exceeded, could cause a degradation of the EDG's capability. This change is designated as more restrictive because new requirements, in the form of a program, are being added to the Technical Specifications.

M.6 The CTS does not contain specific requirements for a Technical Specification Bases Control Program that controls changes to the Bases. ITS 5.5.13 specifies the programmatic controls for processing changes to the Bases of the ITS. This changes the CTS by adding the requirements for the Technical Specification Bases Control Program.

The purpose of ITS 5.5.13 is to establish a means for processing changes to the Bases of the ITS without NRC approval prior to implementation. This change is acceptable because it establishes criteria that allow changes to the Bases without prior NRC approval as long as the change does not require NRC approval pursuant to 10 CFR 50.59. In addition, the program assures consistency with the Technical Specifications and the UFSAR. This change is designated more restrictive because of new requirements, in the form of a program, are being added to the Technical Specifications.

M.7 Regarding lines of authority, CTS 6.2.1.a states, "These requirements shall be documented in the UFSAR." ITS 5.2.1.a states, "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 UFSAR/QA Plan." This changes the CTS by specifying that the plant-specific titles are specified in the QA Plan, as well as the UFSAR.

This change is acceptable because the relationship of the plant-specific titles to the titles used in the Technical Specifications and industry standards is already described in the UFSAR and QA Plan. This change adds this requirement to the Technical Specifications. This change is designated more restrictive because it requires that information be maintained in additional documents.

M.8 The second paragraph of ITS 5.6.2 includes detail to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1.8 does not contain this level of detail. This changes the CTS by requiring additional detail be included in the Annual Radiological Environmental Operating Report.

The purpose of the second paragraph of ITS 5.6.2 is to specify detail to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

M.9 ITS 5.6.6 requires a report be submitted within 14 days after entering Condition B of ITS 3.3.3, PAM Instrumentation. ITS 5.6.6 also states, "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." The CTS do not include these requirements. This changes the CTS by requiring a report to be submitted within 14 days after entering Condition B of ITS 3.3.3 and specifying the contents of the report.

The purpose of ITS 5.6.6 is to ensure that a report is submitted within the following 14 days after entering Condition B of ITS 3.3.3, and that it includes the required information. This change is acceptable because it provides guidance on the reporting requirements for Post Accident Monitoring. This change is designated more restrictive because it adds a new reporting requirement to the Technical Specifications.

M.10 CTS 6.9.1.9, "Annual Radiological Effluent Release Report," states, "A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station…" The ITS 5.6.3 Note replaces the word "should" with "shall." This changes the CTS by clarifying that when a single submittal is made for a multiple unit station, sections common to all units are to be combined.

This change is acceptable because it makes the portions of the Annual Radiological Effluent Release Report common to both units consistent. This change is designated more restrictive because it changes a recommendation for what is to be included in a report to a requirement.

M.11 CTS 3.11.2.5, Explosive Gas Mixture, limits the concentration of oxygen allowed in the waste gas decay tanks. CTS 3.11.2.6, Gas Storage Tanks, limits the quantity of radioactivity contained in each gas storage tank. CTS 3.11.1.4 limits the quantity of radioactive material contained in each of the specified unprotected outdoor tanks. ITS 5.5.11, Explosive Gas and Storage Tank Radioactivity Monitoring Program, include limits on hydrogen in addition to oxygen in the waste gas decay tanks, and requires the program address requirements specified in ITS 5.5.11. This changes the CTS by requiring a new program and specifying certain requirements the program must meet. Changes moving Actions and Surveillance Requirements to the TRM are addressed by DOC LA.7.

The purpose of ITS 5.5.11 is to consolidate the requirements associated with explosive gas and storage tank radioactivity monitoring and specify certain requirements the associated program must meet. This change is acceptable because consolidating and clarifying the requirements provides additional assurance that the requirements will be met. This change is designated as a more restrictive change because a new program and certain requirements are being added to the Technical Specifications.

M.12 CTS 6.2.4.1 states, "The Shift Technical Advisor shall serve in an advisory capacity to Shift Supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit." ITS 5.2.2.f states, "An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift." This changes the CTS by adding more detail to technical areas for which the STA is to provide support, and states that the STA will meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

This change is acceptable because it clarifies STA qualifications. This change is designated more restrictive because it specifies more technical areas the STA must be able to support and requires that the STA meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

M.13 Unit 2 CTS 6.12.1, High Radiation Area, "*," states, "Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas." Unit 2 CTS 6.12.1 applies for control of entry into high radiation areas in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr. Unit 2 CTS 6.12.2 states that the requirements of 6.12.1 also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. ITS 5.7.2, whose applicability is the same as Unit 2 CTS 6.12.2, does not include this allowance. This changes the CTS by deleting the exemption from the RWP issuance requirement for entering the high radiation areas addressed by Unit 2 CTS 6.12.2.

The purpose of the exemption from the RWP issuance requirement for entering the high radiation areas addressed by Unit 2 CTS 6.12.2 is to provide flexibility in performing duties for appropriately qualified personnel. This change is acceptable because for the areas where the intensity of radiation is at the levels addressed by Unit 2 CTS 6.12.2, it is appropriate to use an RWP. This change is designated more restrictive because an exemption from a requirement is being deleted.

M.14 CTS 6.9.1.5.a requires, "A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions." CTS 6.9.1.5.a also states, "In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions." ITS 5.6.1 states, "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 deep dose equivalent (reported in person-rem) according to work and job functions." ITS 5.6.1 also states, "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." This changes the CTS by changing dose and exposure terminology to the more precise deep dose equivalent terms. It also changes the CTS by clarifying that the personnel for whom reporting is done are those for whom monitoring was performed.

This change is acceptable because it provides more precise terminology which is currently in use, and is more specific about who is reported on. This change is designated more restrictive because the requirement is more precise about what is to be reported.

M.15 CTS 6.2.1.d states, "The management position responsible for training of the operating staff and the management position responsible for the quality assurance functions shall have sufficient organizational freedom including sufficient independence from cost and schedule when opposed to safety considerations." CTS 6.2.1.e states, "The management position responsible for health physics shall have direct access to that onsite individual having responsibility for overall facility management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures." ITS 5.2.1.d states, "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." This changes the CTS by stating that specified individuals, not just a particular manager, have sufficient organizational freedom and sufficient independence from operating pressures to perform their work. Also, rather than having access to particular managers, or the authority to cease work for reasons specified in the Specifications, the individuals have sufficient freedom to perform their work.

The purpose of CTS 6.2.1.d and CTS 6.2.1.e is to provide the individuals responsible for training of the operating staff, quality assurance functions, and health physics, with sufficient organizational freedom and independence from operating pressures. This change is acceptable because it provides additional flexibility to individuals with responsibilities for ensuring proper unit operation and proper completion of activities. This change requires the facility to provide the appropriate individuals with the specified flexibility. This change also makes the requirements consistent for people responsible for training of operating staff, quality assurance functions, and health physics functions. This change is designated more restrictive because it requires the facility to provide the additional specified flexibility.

M.16 CTS 6.12.1.b states that one of the optional criteria that allows entry into a high radiation area is, "A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them." ITS 5.7.1.e and ITS 5.7.2.e state, "Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry." This changes the CTS by expanding the requirement to apply to all the options for conditions allowing entry into a high radiation area, and adding the criteria that, "These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry." The phrase, "Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals," is addressed by DOC L.17.

The purpose of the second sentence in CTS 6.12.1.b is to ensure personnel entering high radiation areas are aware of dose rates in the area. This change is acceptable because it provides additional guidance to ensure personnel are aware of the relevant dose rates. This change is designated as more restrictive because additional criteria are added to the requirements for entering a high radiation area.

M.17 One option allowed by CTS 6.12.2 for personnel to enter a high radiation area with radiation intensity greater than 1000 mrem, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates, is to have, "A radiation monitoring device which continuously indicates the radiation dose rate in the area." ITS 5.7.2 does not include this allowance. This changes the CTS by deleting one of the acceptable means for providing personnel radiation exposure information.

This change is acceptable because more comprehensive monitoring is appropriate for entry into areas of such high exposure rates. This change is designated more restrictive because one means of exposure monitoring for a specific criteria is deleted.

M.18 CTS Table 6.2-1 states, "Procedures will be established to insure that NRC policy statement guidelines regarding work hours established for employees are followed." ITS 5.2.2.d states, "Administrative procedures shall be developed and implemented to limit working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel). The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime. Any deviation from the above guidelines shall be authorized by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines is not authorized." This changes the CTS by adding specific requirements for limiting work hours of personnel who perform safety related functions. The change not referencing the NRC policy statement guidelines regarding work hours is discussed in DOC L.24.

The purpose of the CTS Table 6.2-1 statement regarding work hours is to provide guidance limiting work hours of personnel who perform safety related functions. This change is acceptable because it provides specific guidance on who the applicable personnel are, procedural controls, and deviations from the guidance, without a general reference to NRC guidance. This change is designated as more restrictive because it provides more specific direction on work hours of personnel who perform safety related functions.

M.19 As part of one option for equipment required to enter a high radiation area as specified in ITS 5.7.1.d.4 and 5.7.2.d.3, the specifications require, "A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and," one of two other criteria be met for entering a high radiation area. CTS 6.12.c does not include this requirement. This changes the CTS by adding an additional requirement for entering a high radiation area.

This change is acceptable because a self-reading dosimeter provides an additional means by which personnel in a high radiation area can ensure they do not exceed radiation exposure limits. This change is designated as more restrictive because an additional criteria is specified for entering a high radiation area.

M.20 ITS 5.5.15.b states, "The containment design pressure is 45 psig." The CTS does not include such a statement. This changes the CTS by adding a design criterion to the Technical Specifications.

This change is acceptable because the design criteria is already established by the unit design and does not change frequently. This change is designated as more restrictive because an additional design criterion is specified in the Technical Specifications.

M.21 CTS 4.7.8.1 provides ventilation filter testing requirements for the safeguards area ventilation systems (SAVS). Each system is described as having one SAVS exhaust fan and one auxiliary building HEPA filter and charcoal adsorber assembly. ITS

5.5.10 provides ventilation filter testing requirements for the ECCS Pump Room Exhaust Air Cleanup System (PREACS) trains. Each ECCS PREACS train is described as having one safeguards area exhaust fan, one Auxiliary Building Central exhaust system fan, and respective filters, controls, and dampers. This changes CTS by adding additional equipment tested as part of the ventilation filter testing requirements, and changing the testing criteria accordingly, to conform to the system as described in NUREG-1431.

The purpose of ITS 5.5.10 testing criteria for the ECCS PREACS is to provide assurance adequate filtration of the ECCS PREACS exhaust, and that the filtration does not interfere with adequate cooling of ECCS components in the affected areas. All references to filtration and air flow are changed to account for the addition of the Auxiliary Building central exhaust system related components, which are manually actuated. Testing of the auxiliary building HEPA filter and charcoal adsorber assembly is modified to include flow contribution from the Auxiliary Building central exhaust system fans. The system flow rate specified for CTS 4.7.8.1.b.1, 4.7.8.1.d, 4.7.8.1.e, and 4.7.8.1.f is changed to, "Nominal accident flow for a single train actuation." The system flow rate specified for CTS 4.7.8.1.b.3 is changed to, "...one ECCS PREACS train provides greater than the minimum required cooling flow for ECCS equipment." CTS 4.7.8.1.d.1 is changed to state that the flow rate used for testing the pressure drop across the HEPA filter and charcoal adsorber assembly is ≤ 39,200 cfm. A Note is added to CTS 4.7.8.1 that states, "Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber." These changes are acceptable because they add requirements for system components consistent with the intent of NUREG 1431. Specific testing values are changed to properly accommodate these changes in system testing.

References to specific values for testing filter banks, except for pressure drop testing, is replaced with a requirement to perform the test with one train of ECCS PREACS aligned in the post-accident flow configuration. An explanation is added to clarify that flow is acceptable if it is greater than or equal to the minimum required cooling flow for ECCS equipment, and if it has less than the maximum design flow rate of the filter bank (39,200 cfm). The proposed surveillance requirement parameters establish operability of the ventilation system to provide cooling to ECCS equipment and to provide filtration of potential airborne radioactivity prior to being exhausted to the atmosphere. The ECCS PREACS surveillance requirements will ensure that a single train will provide the necessary exhaust flow rate from the ECCS pump rooms. Each ECCS PREACS train includes a HEPA filter and a charcoal adsorber assembly for this purpose. The design (maximum) flow rate for one filter bank is 39,200 cfm, which is based on providing a minimum residence time within the charcoal adsorber. Surveillance requirements will ensure that the flow rate through the filter bank is below the maximum flow rate. Based on testing and engineering evaluation, the maximum pressure drop parameter across the HEPA filter and charcoal adsorber is changed from < 6 inches water gauge to <5 inches water gauge.

These changes are acceptable because they provide additional assurance that the required functions are provided by the ECCS PREACS by adding additional equipment required to be OPERABLE and testing requirements appropriate for the equipment configuration at NAPS. This change is designated as more restrictive because additional equipment and respective acceptance criteria are being added.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 6.8.1.i requires written procedures be established, implemented and maintained covering, "Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the Regulatory Guides to the UFSAR.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.7.1 states, "The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1." CTS Table 5.7-1 contains the limits for component cyclic or transient limits and designs cycle or transient limits. ITS 5.5.5 states, "The components identified in the UFSAR, Section 5.2, are designed and shall be maintained within the cyclic or transient design limits." This changes the CTS by moving the limits specified in Table 5.7-1 to the UFSAR and calling them the cyclic or transient design limits.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to maintain the specified components within the cyclic or transient design limits. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.3 CTS 6.8.4.b, "In-Plant Radiation Monitoring," describes a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. ITS 5.0 does not require such a program. This change moves the requirements of CTS 6.8.4.b to the UFSAR.

The purpose of CTS 6.8.4.b is to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This change is acceptable because it does not affect the health and safety of members of the public. The ITS still requires appropriate post-accident monitoring in accordance with ITS 3.3.3. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive, removal of detail, because information is being relocated from the Technical Specifications.

LA.4 CTS 6.2.3 specifies the function, composition, responsibility, and authority of the Station Nuclear Safety (SNS). ITS 5.2 does not contain this requirement. This changes the CTS by deleting the requirements of CTS 6.2.3 and relocating them to the UFSAR.

The purpose of CTS 6.2.3 is to specify the function, composition, responsibility, and authority of Station Nuclear Safety. This change is acceptable because there are no changes to the current requirements since the requirements are being moved to the UFSAR. Additionally, changes to the UFSAR are controlled in accordance with 10 CFR 50.59. These controls are adequate to assure any change is properly reviewed. This change is designated as a less restrictive, removal of detail, because information is being removed from the Technical Specifications.

LA.5 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.7.7.1 (Control Room Emergency Ventilation System) and 4.7.8.1 (Safeguards Area Ventilation System) specify the Surveillance Requirements and Frequencies for demonstrating OPERABILITY. ITS 5.5.10, "Ventilation Filter Testing Program (VFTP)" does not include some of the Surveillance Requirements and Frequencies specified in the CTS. This changes the CTS by moving these details to the VFTP.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements to perform tests on the ventilation filters in a manner consistent with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, and ANSI N510, 1975. Also, this change is acceptable because these types of procedural details will be adequately controlled in VFTP. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.6 CTS 6.5, 6.6.1.b, 6.8.2, 6.8.3, and 6.15.b specify the function, composition, use of alternates, meeting frequency, quorum, responsibilities, authority, and records of the Station Nuclear Safety and Operating Committee (SNSOC) and the Management Safety Review Committee (MSRC). CTS 6.5 also specifies the use of consultants, reviews and audits for the MSRC. ITS 5.0 does not contain these requirements. This changes the CTS by relocating the requirements for the SNSOC and MSRC to the QA Topical Report in the UFSAR.

The purpose of CTS 6.5, 6.6.1.b, 6.8.2, 6.8.3, and 6.15.b is to specify the function, composition, use of alternates, meeting frequency, quorum, responsibilities, authority, and records of the Station Nuclear Safety and Operating Committee (SNSOC) and the Management Safety Review Committee (MSRC), and the use of consultants, reviews and audits for the MSRC. The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The description of the means by which the SNSOC and MSRC support the Technical Specifications and perform other tasks is moved to the UFSAR. Also, this change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information concerning the SNSOC and MSRC is being relocated from the Technical Specifications.

LA.7 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 3.11.1.4, Liquid Holdup Tanks, imposes limits on the quantity of radioactive material contained in each tank. CTS 3.11.2.5, Explosive Gas Mixture, limits the oxygen concentration in the Waste Gas Decay Tanks to ensure that the concentration of potentially explosive gas mixtures in the Waste Gas Decay Tanks is maintained below the flammability limits for hydrogen and oxygen. CTS 3.11.2.6, Gas Storage Tanks, imposes limits on the quantity of radioactive material contained in each tank. ITS 5.5.11, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," does not contain the specific requirements, Applicability, Actions, and Surveillance Requirements in CTS 3.11.1.4, CTS 3.11.2.5, and CTS 3.11.2.6. This changes the CTS by moving this information to the TRM.

The removal of these details for performing actions and surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the for an Explosive Gas and Storage Tank Radioactivity Monitoring Program. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Technical Requirements Manual. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirementsare being removed from the Technical Specifications.

LA.8 *(Type 3 – Removing Procedural Details for Meeting TS Requirements)* CTS 6.8.4.g contains the requirements for the Configuration Risk Management Program. ITS 5.0 does not include requirements for the Configuration Risk Management Program. This changes the CTS by moving the requirements for the Configuration Risk Management Program to the UFSAR.

The removal of these details for assessing risk in relation to equipment inoperability and performing related actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the risk informed Required Actions and the definition of OPERABILITY for the related equipment. Also, this change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.9 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 6.9.1.7.e specifies the revisions and dates of the referenced methodologies, and the LCOs for which the referenced methodologies are used. ITS 5.6.5.b does not contain this level of detail. This changes the CTS by moving the specific methodology references for revisions, dates, and LCOs to the COLR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the references for the COLR and only NRC-approved methodologies may be used. Also, this change is acceptable because the removed information will be adequately controlled in the COLR. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.10 CTS 6.8.4.f, "Radiological Environmental Monitoring Program," describes a program to monitor the radiation and radionuclides in the environs of the plant. ITS 5.0 does not require such a program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the ODCM.

The purpose of CTS 6.8.4.f is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program. The removal of the requirement for this program from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.6.2 still requires an annual report of the results of the "Radiological Environmental Monitoring Program." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. This change is designated as a less restrictive, removal of detail, because the requirements for a program are being removed from the Technical Specifications.

LA.11 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.4.10.1.2 states, "In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval." ITS 5.5.7, "Inservice Testing Program," specifies the controls for inservice testing of ASME Code Class 1, 2, and 3 components. This changes the CTS by moving these requirements to the Inservice Testing Program.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements for an Inservice Testing Program. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ISI/IST Program. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 CTS Table 6.2-1 specifies that the shift crew may be one less than the minimum complement, except for the Shift Supervisor, for a period of time not to exceed 2 hours. CTS Table 6.2-1 also takes an exception that the provision for being less than minimum shift crew complement does not apply for any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. ITS 5.2.2.b does not make these exceptions to the requirements of 10 CFR 50.54

(m)(2)(i). This changes the CTS by allowing shift crew composition to be less than the manning requirements without specifying exceptions to this allowance.

The purpose of the allowance to have less than the required the shift crew manning requirements is to accommodate short term unexpected absences of shift crew personnel. This change is acceptable because 10 CFR 50.54 (m)(2)(ii) still requires a minimum of two SROs and four ROs when the shift crew composition is less than the manning requirements, which is enough to safely operate the unit. This change is designated less restrictive because restrictions regarding shift manning are being deleted from the CTS.

L.2 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.5.c states the contents of an annual report to be submitted to the Nuclear Regulatory Commission which contains the results of specific activity analyses in which the primary coolant exceeded the limits of the RCS Specific Activity Specification. ITS 5.6 does not contain any requirements for such an annual report. This changes the CTS by not including the requirements for the annual report of specific activity analyses in which the primary coolant exceeded the limits of the RCS Specific Activity Specification.

The purpose of CTS 6.9.1.5.c is to specify the requirements for submitting an annual report which contains the results of specific activity analyses in which the primary coolant exceeded the limits of the RCS Specific Activity Specification. This change is acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported by 10 CFR 50.73. Subsequent reports would be provided as needed, without requiring a specific annual report. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.3 CTS 6.2.2 states, "The Facility organization shall be as shown in the UFSAR." ITS 5.2.2 states, "The Facility organization shall include..." and describes the facility organization. This changes the CTS by deleting the requirement to have the description of the facility organization in the UFSAR.

The purpose of CTS 6.2.2 is to provide guidance on what the Facility organization should be. This change is acceptable because ITS 5.2.2 and 10 CFR 50.54(m)(2)(i) continue to identify minimum unit operations shift manning and other plant manning requirements, but the remainder of the facility organization does not need to be referenced in the UFSAR. This change is designated as less restrictive because it does not require that the facility organization be shown in the UFSAR.

L.4 CTS 6.1.1 states, "The Site Vice President shall be responsible for overall facility operation. In his absence, the Manager - Station Operations and Maintenance shall be responsible for overall facility operation. During the absence of both, the Site Vice-President shall delegate in writing the succession to this responsibility." ITS 5.1.1

states, "The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence." This changes the CTS by not specifying the title of the person with responsibility for overall facility operation, and allowing the plant manager to delegate the responsibility to someone other than the Manager - Station Operations and Maintenance if that person is not absent.

The purpose of CTS 6.1.1 is to provide a means for specifying the person with responsibility for overall plant operation. This change is acceptable because it identifies the generic title of the position with the specified responsibility, and requires the responsibility be delegated in writing. The responsibility for overall unit operation is still clearly identified in writing, but is less proscriptive about to whom the responsibility is delegated. This change is designated less restrictive because the plant specific title of the person with the responsibility is not specified, and the second person in the succession to this responsibility is not mandated.

L.5 CTS 6.1.2 states, "A management directive to this effect, signed by the Senior Vice President-Nuclear, shall be issued to all station personnel on an annual basis," regarding delegation of the control room command function. ITS 5.1.2 does not include such a requirement. This changes the CTS by deleting the requirement to issue a management directive annually.

The purpose of CTS 6.1.2 is to specify the plant specific means of implementing the NUREG-0737 requirement to notify employees of shift supervisor responsibilities. This change is acceptable because the NUREG-0737 requirement is not changed and the plant specific implementation of the requirement is not appropriate for the Technical Specifications. This change is designated as a less restrictive change because a required action is removed from the Technical Specifications.

L.6 CTS 6.2.1.b states, "The Site Vice President shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant." CTS 6.2.1.c states, "The Vice President – Nuclear Operations 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." CTS 6.15 states, "Changes to the ODCM:... b. Shall become effective after...the approval of the Site Vice President." ITS 5.2.1.b substitutes "plant manager" for "Site Vice President," ITS 5.2.1.c substitutes "A specified corporate officer" for "The Vice President – Nuclear Operations," and ITS 5.5.1.b substitutes "plant manager" for "Site Vice President." This changes the CTS by using less specific designations for the positions with the respective responsibilities.

These changes are acceptable because the responsibilities remain the same, but allow other documents to identify the plant-specific titles associated with the generic titles.

This change is designated less restrictive because specific titles associated responsibilities are deleted from the Technical Specifications.

L.7 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3, "Startup Reports," contains requirements for submitting a report following receipt of an operating license; installation of fuel that has a different design or has been manufactured by a different fuel supplier; modifications that may have altered the nuclear, thermal, or hydraulic performance of the unit; and amendments to the license involving planned increase in power operation. The ITS does not contain such reporting requirements. This changes the CTS by deleting the requirements of CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3.

The purpose of CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3, is to provide a summary of plant startup and power escalation testing following the four specified conditions as verification that the unit operated as expected. This change is acceptable because the regulations provide adequate reporting requirements. If there were any unit conditions outside the expected parameters during unit startup, they would be reported to the NRC if they met the reporting requirements in the regulations. Otherwise, the reports would document that the unit operated as expected and already approved by the NRC, as required by regulations. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.8 CTS Table 6.2-1 includes requirements on SS, SRO, RO, AO, and STA position manning for each unit that are beyond what is required by 10 CFR 50.54(m)(2)(i). The ITS does not include these conditions. This changes the CTS by deleting certain criteria regarding how manning is distributed.

The intent of the conditions placed on unit staff manning is to state management policies regarding how the required positions are distributed between the two units at the site. This change is acceptable because this distribution can still be retained in accordance with management policy, but does not need to be retained in the ITS. The 10 CFR 50.54(m)(2)(i) requirements for staff manning are still required to be met. This change is designated less restrictive because conditions regarding the required staff manning are being deleted.

L.9 CTS Table 6.2-1 requires that with both units in MODE 5 or 6 or defueled, two Auxiliary Operators (AOs) be part of the staff manning, one AO assigned to each unit. ITS 5.2.2.a states, "A total of four non-licensed operators shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A non-licensed operator, who may be one of the four assigned to a control room, shall be assigned to each reactor containing fuel." This changes the CTS by only requiring one AO for the control room for each reactor containing fuel rather than two, not requiring an AO for a defueled reactor, and not making the AO requirement dependent on the status of the opposite unit.

The purpose of the AO requirements in CTS Table 6.2-1 is to provide assurance that sufficient AOs are on the shift crew. This change is acceptable because it still provides at least one AO for each reactor containing fuel. It also makes the AO requirement more unit specific, designating the number of AOs required based on unit condition, and not requiring an AO for a defueled unit. This change is designated less restrictive because unit AO manning is reduced and not dependent on the status of the opposite unit.

L.10 CTS Table 6.2-1, with regard to work hour procedures, states, "In addition, procedures will provide for documentation of authorized deviations from these guidelines and that the documentation is available for NRC review." ITS 5.0 does not include such a requirement. This changes the CTS by deleting a requirement to have a procedure for documentation of authorized deviations from the work hour guidelines and to have the documentation available for NRC review.

The purpose of the CTS Table 6.2-1 requirements regarding a procedure for documentation of authorized deviations from the work hour guidelines and to have the documentation available for NRC review is to assist in documenting and correcting guideline deviations. This change is acceptable because the work hour guidelines are still required to be met, but retaining the requirements for procedural documentation of authorized deviations will be retained under licensee control. This change is designated less restrictive because a requirement for procedural controls is being deleted from the CTS.

L.11 CTS 6.2.2.c references requirements for a health physics technician. CTS 6.12.1, footnote "*" describes a Health Physics technician allowance. CTS 6.12.2 references a responsibility of the Shift Supervisor on duty and/or the Plant Health Physicist. ITS 5.2.2.d references a radiation protection technician, and ITS 5.7.1 references Radiation Protection personnel, and ITS 5.7.2 references the radiation protection shift supervisor, radiation protection manager or his or her designee responsibilities, respectively. This changes the CTS by changing the titles of the personnel in the specified positions to more generic titles.

The purpose of these CTS requirements is to specify qualifications of people assigned particular duties. This change is acceptable because though the titles are more generic, as described in ANSI/ANS 3.1, the personnel will continue to be qualified to plant standards. These changes are designated as less restrictive because the titles of personnel in designated positions are less specific.

L.12 CTS 6.8.4 states that one of the programs to be established, implemented, and maintained is, "A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels." ITS 5.5.2 requires that the program minimize the same leakage. This changes the CTS by requiring the program provide controls to minimize instead of reduce leakage.

The purpose of the CTS 6.8.4 program for leakage from primary coolant sources outside containment is to provide guidance for the program which would reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This change is acceptable because the program will still keep the potential leakage to as low as practical levels, but will require leakage be minimized rather than reduced, which is more commensurate with the term as low as practical. This change is designated as less restrictive because a less stringent requirement is being applied to a program.

L.13 ITS 5.7.2.f states, "Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device." CTS 6.12.2 does not include such an allowance. This changes the CTS by providing an additional method by which to control a high radiation area meeting the criteria of 5.7.2.

The purpose of ITS 5.7.2.f is to provide an adequate optional means of controlling access to a high radiation area described in ITS 5.7.2. This change is acceptable because it provides adequate controls for the areas addressed by ITS 5.7.2 without requiring controls for a much larger area that would restrict work and access while providing no substantial improvement in control of the area. This change is designated as less restrictive because it provides an additional option for controlling the areas addressed by ITS 5.7.2.

L.14 *(Category 2 – Relaxation of Applicability)* ITS 5.5.14 provides criteria for the Safety Function Determination Program (SFDP), as referenced in ITS LCO 3.0.6. This provides an exception to ITS LCO 3.0.2 when a supported system LCO is not met solely due to a support system LCO not being met, such that the Conditions and Required Actions associated with this supported system are not required to be entered and there has been no loss of safety function . The CTS do not include such an exception to CTS LCO 3.0.2. This changes the CTS by including the criteria for an exception to CTS LCO 3.0.2.

The purpose of ITS 5.5.14 is to allow an exception to ITS LCO 3.0.2. The exception is allowed when a supported system LCO is not met solely due to a support system LCO not being met, such that the Conditions and Required Actions associated with this supported system are not required to be entered, and there has been no loss of safety function. This change is acceptable because the requirements continue to ensure that the supported system process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The support system LCO Conditions must still be entered, and the SFDP includes criteria to determine when the supported system LCO Conditions must be entered. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.15 CTS 6.9.1.4 requires annual reports described in CTS 6.9.1.5 be submitted prior to March 1 of each year. ITS 5.6.1 requires the Occupational Radiation Exposure Report to be submitted by April 30 of each year. This changes the CTS by allowing an additional 2 months to submit the Occupational Radiation Exposure Report each year.

The purpose of the due date for submitting the Occupational Radiation Exposure Report is to ensure it is provided in a reasonable period of time to the NRC for review. This change is acceptable because the report is still required to be submitted in a reasonable time frame. The change makes the due date consistent with the due dates for ITS 5.6.2 (Annual Radiological Environmental Operating Report) and ITS 5.6.3 (Radioactive Effluent Release Report). This change is designated as less restrictive because it allows more time to prepare and submit an annual report to the NRC.

L.16 CTS 6.12.1 states for high radiation areas, "...entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit." ITS 5.7.1.b and ITS 5.7.2.b state for high radiation areas, "Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures." This changes the CTS by allowing an equivalent document to be used for access control. The addition of details required in the RWP is addressed by DOC M.4.

The purpose of the specified phrase in CTS 6.12.1 is to designate the document through which access is controlled to the specified high radiation areas. This change is acceptable because a proper document is still required, but it may serve the same purpose as an RWP without having to be specifically called an RWP. This change is designated a less restrictive because an alternate document may be used for access control in lieu of an RWP.

L.17 Unit 1 CTS 6.12, High Radiation Area, footnote "*," states, "Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas." Unit 2 CTS 6.12, High Radiation Area, footnote "*," states, "Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas." ITS 5.7.1 states, "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." This changes the CTS by allowing Unit 1 personnel other than the Health Physics personnel to be exempt from the RWP issuance requirement, and for both Unit 1 and Unit 2, the personnel are allowed to use the exemption for the performance of assigned duties, not only radiation protection duties. These criteria apply in high radiation areas with exposure rates <1000 mrem/hr. Changing the term "Health Physics" to "radiation protection" is addressed by DOC L.11.

The purpose of CTS 6.12 footnote "*" is to provide an allowance for qualified personnel to not have to issue an RWP during the performance of their assigned radiation protection duties. This is because their training and the use of approved radiation protection procedures provides assurance that their personnel exposure will be within established limits. This change is acceptable because the escort of people by these trained individuals for any duties, not just for radiation protection, using approved radiation protection procedures, also provides assurance that the personnel exposure of the of the people being escorted will be within established limits. These changes are designated as less restrictive because a larger group of individuals will be eligible to be exempt from RWP issuance, and for a wider variety of duties.

L.18 CTS Table 6.2-1 states the qualifications for the person that assumes the control room command function during the absence of the Shift Supervisor, and excludes the STA as a person who can assume that function. ITS 5.1.2 does not include this exclusion of the STA. This changes the CTS by allowing an STA that holds a valid SRO license to assume the control room command function during the absence of the Shift Supervisor.

The purpose of the exclusion of the STA from being allowed to assume the control room command function during the absence of the Shift Supervisor is to provide assurance that the independent STA function is retained. This change is acceptable because the STA assuming the control room command function is a temporary state, and if the STA is qualified to assume the function, the STA is trained to man that position. There is no such explicit exclusion regarding manning in 10 CFR 50.54(m)(2)(i). This change is designated as a less restrictive change because an additional member of the shift crew composition is allowed to assume the control room command function during the absence of the Shift Supervisor.

L.19 CTS 6.4.1 states, "The Manager – Nuclear Training is responsible for ensuring that retraining and replacement training programs for the licensed facility staff meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). Also, a retraining and replacement training program for non-licensed facility staff shall meet or exceed the recommendations of Section 5 of ANS 3.1 (12/79 Draft)*." CTS 6.4.1 footnote "*" states, "Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase."" ITS 5.0 does not include these requirements. This changes the CTS by not specifying who is responsible for ensuring the requirements of 10 CFR 55.59(c) and 55.31(a)(4) are met, and not specifying requirements for non-licensed facility staff training.

The purpose of CTS 6.4.1 is to assign responsibility for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and also to specify criteria for the training of the non-licensed facility staff. This change is acceptable because the requirements of 10 CFR 55.59(c) and 55.31(a)(4) are still required to be met, and the training of non-licensed facility staff is still expected to meet appropriate standards. Identification of the person responsible for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and specification of the training requirements to be met for non-licensed facility staff is addressed by plant processes. This change is designated as less restrictive because designation of the responsibility for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and requirements for the training program for non-licensed facility staff are deleted.

L.20 *(Category 7 – Relaxation Of Surveillance Frequency)* ITS 5.5.7.c states, "The provisions of SR 3.0.3 are applicable to inservice testing activities." CTS does not include an equivalent statement. This changes the CTS by allowing 24 hours or up to the limit of the Frequency, whichever is less, to perform inservice testing if it is discovered that the inservice testing requirements were not performed, instead of declaring the component inoperable.

The purpose of ITS 5.5.7.c is to make the significance of the inservice testing requirements the same as the Surveillance Testing requirements. This change is acceptable because the new Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The inservice testing Frequencies are still required to be met, except when the criteria are met to allow an extension of 24 hours or up to the limit of the Frequency. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.21 ITS 5.6.1 allows dose assignments to various duty functions to be estimated using, among other things, an electronic dosimeter. CTS 6.9.1.5 does not include this allowance. This changes the CTS by including an electronic dosimeter as one of the ways by which dose assignments to various duty functions may be estimated.

The allowance in ITS 5.6.1 to use an electronic dosimeter as one of the ways by which dose assignments to various duty functions may be estimated allows the use of equipment that can provide valid measurements for this requirement. This change is acceptable because measurements from the electronic dosimeters will be used in conjunction with other equipment to make a best estimate. This change is designated less restrictive because it allows the use of equipment not previously allowed for performing a required estimate.

L.22 Unit 1 CTS 4.4.5 Table 4.4-1 states that if an additional steam generator is in category C-3, one Action Required is, "Report to NRC & obtain approval prior to operation." ITS Table 5.5.8-2 for the same condition states, "Report to NRC pursuant to 5.6.7.c." This changes the CTS by not requiring obtaining NRC approval prior to operation in the event an additional steam generator is found to be in the category C-3.

The purpose of the reporting requirements of CTS 4.4.5 is to ensure the NRC is informed of steam generator tube inspection results which fall into the category C-3. CTS 4.4.5.5 and ITS 5.6.7 require results of steam generator tube inspections which fall into Category C-3 result in prompt notification of the NRC pursuant to 10 CFR 50.72 and 10 CFR 50.73. This change is acceptable because the regulations provide adequate reporting requirements. Prompt reporting to the NRC is still required in accordance with regulations. The NRC may still require the licensee to obtain approval prior to unit operation, but the requirement will not be specified in the Technical Specifications. This change is designated as less restrictive because it does not require input from the NRC before resuming unit operation.

L.23 CTS 6.12.2 states, regarding areas in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates, "In addition, locked doors shall be provided to prevent unauthorized entry into such areas..." ITS 5.7.2 states, "...areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry." This changes the CTS by allowing the doors to be guarded as an option to locking them.

The purpose of CTS 6.12.2 with regard to preventing unauthorized access is to state the means by which to prevent such entry. This change is acceptable because adequate controls are maintained to prevent unauthorized access, while allowing reasonable flexibility regarding how to establish those controls. These changes are designated as less restrictive because it allows an additional means of preventing unauthorized entry into the specified high radiation area.

L.24 CTS Table 6.2-1 states, "Procedures will be established to insure that NRC policy statement guidelines regarding work hours established for employees are followed." ITS 5.2.2.d states, "Administrative procedures shall be developed and implemented to limit working hours of personnel who perform safety related functions..." This changes the CTS by not referencing the NRC policy statement guidelines regarding work hours as the source of guidance for limiting work hours.

The purpose of the work hour control note in CTS Table 6.2-1 is to provide reasonable assurance that impaired performance caused by excessive work hours will not jeopardize safe plant operation. This change is acceptable because specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision making process to minimize the potential for impaired personnel performance, and that established procedure control processes will provide sufficient controls for changes to that procedure. Referencing the NRC policy statement guidelines regarding work hours is not required to accomplish this. This change is designated as a less restrictive change because the NRC policy statement guidelines regarding work hours is not specifically referenced in the Technical Specifications.

L.25 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 6.8.4.a.5 requires, "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." ITS 5.5.4 states, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency." CTS does not include this provision. This changes the CTS by permitting a 25% extension of the interval specified in the Frequency.

The purpose of reporting the cumulative and projected dose contributions from radioactive effluents is to routinely evaluate cumulative and projected dose contributions from radioactive effluents. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. This change increase the time allowed to submit the report of projected dose contributions from radioactive effluents and is acceptable because the change will have no effect on the outcome of the calculations, and the reports will still be provided in a timely basis. This change is designated as less restrictive because more time is provided to submit the report under the ITS than under the CTS.

L.26 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.6 states, "Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis..." ITS 5.6.4 states, "Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis..." This changes the CTS by deleting the requirement to include documentation of all challenges to the Reactor Coolant System PORVs or safety valves in the monthly report.

The purpose of CTS 6.9.1.6 is to ensure the NRC receives appropriate routine reports of operating statistics and shutdown experience on a monthly basis. This change is acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. The change deletes the requirement to include documentation of all challenges to the Reactor Coolant System PORVs or safety valves in the monthly report, though they are still required in the annual report. The guidance of NUREG 0694, "TMI-Related Requirements for New Operating

Licenses," states, "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.27 ITS 5.7.1.4.d.3 states that one of the options for devices an individual or group shall possess for radiation monitoring when entering a high radiation area with a dose rate not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation is, "A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area." ITS 5.7.2.4.d.2 states that one of the options for devices an individual or group shall possess when entering a high radiation area with a dose rate exceeding 1.0 rem/hour at 30 Centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or any surface penetrated by the radiation is, "A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area." CTS 6.12.1 and 6.12.2 do not contain these options for an individual or group. This changes the CTS by providing an additional device an individual entering these high radiation areas must possess for radiation monitoring.

The purpose of ITS 5.7.1.4.d.3 and ITS 5.7.2.4.d.2 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose of personnel in high radiation areas has been provided.

L.28 CTS 6.12.1.b states that one of the optional criteria that allow entry into a high radiation area is, "An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit." ITS 5.7.1.d.4 states, "A self reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and, (i) be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or (ii) be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance." ITS 5.7.2.d.3 reads the same as ITS 5.7.1.d.4, except the last phrase, "communicate with individuals in the area who are covered by such surveillance," is replaced with the phrase, "communicate with and control every individual in the area." This changes the CTS by deleting the discussion of positive controls over activities and performing radiation surveillances with a requirement for the monitoring device to have continuous dose rate displays and the responsibility to control dose rates in the area, and an option to perform the monitoring of personnel remotely using the specified equipment and processes.

The purpose of 6.12.1.c is to provide the option of monitoring the exposure of individuals in high radiation areas by a separate individual qualified in radiation procedures. This change is acceptable because it provides adequate means of monitoring the personnel in the high radiation areas, but provides added flexibility for how to do it. This change is designated as less restrictive because additional methods for monitoring personnel exposure are provided.

L.29 ITS 5.7.2.4.d.4 states that one of the options for devices that an individual or group shall possess when entering a high radiation area with a dose rate exceeding 1.0 rem/hour at 30 Centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or any surface penetrated by the radiation is, "In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area." CTS 6.12.1 and 6.12.2 do not contain these options for an individual or group. This changes the CTS by providing an additional option for devices an individual entering these high radiation areas must possess.

The purpose of ITS 5.7.2.4.d.4 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose of personnel in high radiation areas has been provided.

L.30 CTS 6.8.2 states, "Each new procedure of 6.8.1 above, except 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved by the SNSOC prior to implementation as set forth in administrative procedures. Procedures of 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved as set forth in the facility's Security Plan, Emergency Plan, and section 6.5.1.6.m of the Technical Specifications, respectively." CTS 6.8.1.d is Security Program implementation. CTS 6.8.1.e is Emergency Plan implementation. CTS 6.8.1.f is Fire Protection Program Implementation. CTS 6.8.3 states, "Procedure changes that require a safety evaluation shall also be reviewed and approved by SNSOC. All other changes shall be independently reviewed and approved as programmatically discussed in the Updated Final Safety Analysis Report." ITS 5.0 does not include statements like those in CTS 6.8.2 and 6.8.3 regarding review and approval of procedures of CTS 6.8.1.d, 6.8.1.e, 6.8.1.f, and review and approval of changes as described in the UFSAR. This changes the CTS by not specifying how these procedures are reviewed and approved.

The purpose of the portions of CTS 6.8.2 and 6.8.3 of concern is to provide assurance that the referenced procedures are changed in accordance with the specified documents. This change is acceptable because the LCO requirements continue to ensure that the appropriate programs are maintained consistent with the licensing basis. The change deletes the requirement that changes to the specified procedures be reviewed and approved as stated in the referenced documents. Procedure changes are conducted in accordance with plant procedures. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.31 CTS 6.8.4.e.5 states that the radioactive effluent control program shall include "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." ITS 5.5.4.e states that the radioactive effluent control program shall include "Determination of cumulative 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology and parameters in the ODCM at least every 31 days." This changes the CTS by not requiring that a projection of the dose contribution for the current calendar quarter and the current calendar year be performed every 31 days.

The purpose of the portions of CTS 6.8.4.e.5 is to determine the cumulative dose contributions for the current calendar quarter and current calendar year and to then project the dose contributions in the future. This is necessary to assess current and future compliance with offsite dose limits. This change is acceptable because the requirements continue to ensure that the appropriate programs are maintained consistent with the licensing basis. The current wording could be construed to require projection for the current quarter and current year. This misleading wording was promulgated in Generic Letter 89-01. The NRC has agreed that the proposed wording represents the intent of the requirements in their approval of TSTF-308, Revision 1. This change is designated as less restrictive because less stringent requirements are being applied in the ITS than were applied in the CTS.

L.32 CTS 1.22 describes the Process Control Program (PCP). CTS 6.14 (Unit 1) and CTS 6.13 (Unit 2) specifies the change control for the PCP. CTS 6.8.1.g requires written procedures be established, implemented, and maintained to cover PCP implementation.

The ITS does not specify requirements for the PCP. This changes the CTS by removing the requirements associated with the contents and maintenance of the PCP.

The purpose of CTS 1.22, CTS 6.14 (Unit 1), CTS 6.13 (Unit 2), and 6.8.1.g is to describe requirements for the PCP in order to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste. This change is acceptable because the requirements for the PCP change control are not required to be in the ITS to provide adequate protection of the public health and safety. Compliance with the specified requirements governing the disposal of radioactive waste is still required. This change is designated as less restrictive because the specific manner in which regulations are being met is being removed from the Technical Specifications.

L.33 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.7.7.2.c states that the relative humidity at which the laboratory test samples of the charcoal adsorber are tested is 95%. ITS 5.5.10.c states that the relative humidity at which the laboratory test samples of the charcoal adsorber are tested is 70%. This changes the CTS by relaxing the criteria for the test of the charcoal adsorber to a 70% humidity level instead of 95%.

The purpose of ITS 5.5.10.c is to verify the charcoal adsorbers can perform their function under the condition assumed in case of a DBA. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Engineering testing and analysis has determined that the maximum relative humidity for the required charcoal adsorber inlet air at North Anna during accident conditions is 70%. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

GENERIC NSHCs

10 CFR 50.92 EVALUATION
FOR
ADMINISTRATIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications with no change in intent. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

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

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR MORE RESTRICTIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional commitments that decrease allowed outage times, increase the frequency of surveillances, impose additional surveillances, increase the scope of specifications to include additional plant equipment, increase the applicability of specifications, or provide additional actions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
RELOCATED SPECIFICATIONS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relocating existing Technical Specification LCOs to licensee controlled documents.

The the Company has evaluated the current Technical Specifications using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the Improved Technical Specifications (ITS) submittal. These specifications have been relocated from the current Technical Specifications to the Technical Requirements Manual.

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

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the North Anna Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to the Technical Requirements Manual, which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR.50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change and any future changes to these requirements will be evaluated per 10 CFR 50.59.

NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety which can be evaluated. However, since the proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431 issued by the NRC, revising the Technical Specifications to reflect the approved level of detail gives assurance that this relocation does not result in a significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES - REMOVED DETAIL

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve moving details out of the Technical Specifications and into the Technical Specifications Bases, the UFSAR, the TRM or other documents under regulatory control such as the Quality Assurance Program Topical Report. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1431 for format and content.

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

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

   The proposed change relocates certain details from the Technical Specifications to other documents under regulatory control. The Bases, UFSAR, and Technical Requirement Manual will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the Technical Specifications. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by Technical Specifications or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no effect on any safety analysis assumptions. In addition, the details to be moved from the Technical Specifications to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431, issued by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail, which indicates that there is no significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 1 RELAXATION OF LCO REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 2 RELAXATION OF APPLICABILITY

The North Anna Nuclear Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the applicability of current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as, "all MODES" or "any operating MODE." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" or "any operating MODE," replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analysis and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1431 and has been evaluated to not be detrimental to plant safety.

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

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

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 3
RELAXATION OF COMPLETION TIME

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies times for completing Required Actions of the associated TS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times (referred to as Allowed Outage Times (AOTs) in the CTS). These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing AOT. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 4
RELAXATION OF REQUIRED ACTION

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from the ISTS is acceptable because the Required Actions take into account the operability status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve deletion of Surveillance Requirements in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify Operability under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which, when combined, include Operability verification of all TS required components for the features specified in the CTS. Adopting this format preference in the ISTS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes which provide exceptions to Surveillance Requirements to provide for variations which do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 7 RELAXATION OF SURVEILLANCE FREQUENCY

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing surveillance testing. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other Technical Specification requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced and; in turn, reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be Operable and capable of performing any accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF REPORTING REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the deletion of requirements in the current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

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

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

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator to any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change does not involve a significant reduction in a margin of safety.

This proposed Technical Specification change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements,

(i) proposed change involves No Significant Hazards Considerations (refer to the Determination of No Significant Hazards Considerations section of this Technical Specification Change Request);

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental affect statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change of this request.

CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

SPECIFIC NSHCs

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.1

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.1 CTS Table 6.2-1 specifies that the shift crew may be one less than the minimum complement, except for the Shift Supervisor, for a period of time not to exceed 2 hours. CTS Table 6.2-1 also takes an exception that the provision for being less than minimum shift crew complement does not apply for any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. ITS 5.2.2.c does not make these exceptions to the requirements of 10 CFR 50.54 (m)(2)(i). This changes the CTS by allowing shift crew composition to be less than the manning requirements without specifying exceptions to this allowance.

The purpose of the allowance to have less than the required the shift crew manning requirements is to accommodate short term unexpected absences of shift crew personnel. This change is acceptable because 10 CFR 50.54 (m)(2)(ii) still requires a minimum of two SROs and four ROs when the shift crew composition is less than the manning requirements, which is enough to safely operate the unit. This change is designated less restrictive because restrictions regarding shift manning are being deleted from the CTS.

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

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

This change allows shift crew composition to be less than the manning requirements for up to two hours without specifying exceptions to this allowance. Shift crew composition is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Shift manning does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows shift crew composition to be less than the manning requirements for up to two hours without specifying exceptions to this allowance. The ITS requirements are considered adequate for shift manning, and temporary reduction in manning will not significantly affect the unit staff's ability to respond to an accident. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.3

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.3 CTS 6.2.2 states, "The Facility organization shall be as shown in the UFSAR." ITS 5.2.2 states, "The Facility organization shall include…" and describes the facility organization. This changes the CTS by deleting the requirement to have the description of the facility organization in the UFSAR.

The purpose of CTS 6.2.2 is to provide guidance on what the Facility organization should be. This change is acceptable because ITS 5.2.2 and 10 CFR 50.54(m)(2)(i) continue to identify minimum unit operations shift manning and other plant manning requirements, but the remainder of the facility organization does not need to be referenced in the UFSAR. This change is designated as less restrictive because it does not require that the facility organization be shown in the UFSAR.

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

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

   This change deletes the requirement to have the description of the facility organization in the UFSAR. Facility organization is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Facility organization does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change deletes the requirement to have the description of the facility organization in the UFSAR. The ITS requirements are considered adequate to provide assurance of adequate shift manning and for responsibility for overall facility operation in that qualified personnel continue to be responsible for facility operation. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.4

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.4 CTS 6.1.1 states, "The Site Vice President shall be responsible for overall facility operation. In his absence, the Manager - Station Operations and Maintenance shall be responsible for overall facility operation. During the absence of both, the Site Vice-President shall delegate in writing the succession to this responsibility." ITS 5.1.1 states, "The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence." This changes the CTS by not specifying the title of the person with responsibility for overall facility operation, and allowing the plant manager to delegate the responsibility to someone other than the Manager - Station Operations and Maintenance if that person is not absent.

The purpose of CTS 6.1.1 is to provide a means for specifying the person with responsibility for overall plant operation. This change is acceptable because it identifies the generic title of the position with the specified responsibility, and requires the responsibility be delegated in writing. The responsibility for overall unit operation is still clearly identified in writing, but is less proscriptive about to whom the responsibility is delegated. This change is designated less restrictive because the plant specific title of the person with the responsibility is not specified, and the second person in the succession to this responsibility is not mandated.

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

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

   This change does not specify the company title of the person with responsibility for overall facility operation, and allows the plant manager to delegate the responsibility to someone other than the Manager - Station Operations and Maintenance if that person is not absent. The company title of the person with responsibility for overall facility operation, and delegation that responsibility is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The company title of the person with responsibility for overall facility operation, and method for delegating that responsibility, do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not specify the company title of the person with responsibility for overall facility operation, and allows the plant manager to delegate the responsibility to someone other than the Manager - Station Operations and Maintenance if that person is not absent. The ITS requirements are considered adequate for responsibility for overall facility operation in that qualified personnel continue to be responsible for facility operation. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.5

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.5 CTS 6.1.2 states, "A management directive to this effect, signed by the Senior Vice President-Nuclear, shall be issued to all station personnel on an annual basis," regarding delegation of the control room command function. ITS 5.1.2 does not include such a requirement. This changes the CTS by deleting the requirement to have such a management directive.

The purpose of CTS 6.1.2 is to specify the plant specific means of implementing the Technical Specification requirements on SS responsibilities. This change is acceptable because the Technical Specification requirement remains the same, and plant specific implementation details are inappropriate for the Technical Specifications. This change is designated as a less restrictive change because a required action is removed from the Technical Specifications.

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

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

This change deletes the requirement for a management directive, signed by the Senior Vice President-Nuclear, to be issued to all station personnel on an annual basis regarding delegation of the control room command function. The requirement for such a management directive is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Issuance of a management directive does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change deletes the requirement for a management directive, signed by the Senior Vice President-Nuclear, to be issued to all station personnel on an annual basis regarding delegation of the control room command function. The ITS requirements are considered adequate for the control room command function because shift manning requirements continue to provide adequate shift coverage, and the process by which the control room command function is delegated can be addressed adequately outside of the Technical Specifications. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.6

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.6 CTS 6.2.1.b states, “The Site Vice President shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant." CTS 6.2.1.c states, "The Vice President – Nuclear Operations 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." CTS 6.15 states, "Changes to the ODCM:... b. Shall become effective after...the approval of the Site Vice President." ITS 5.2.1.b substitutes "plant manager" for "Site Vice President," ITS 5.2.1.c substitutes "A specified corporate officer" for "The Vice President – Nuclear Operations," and ITS 5.5.1.b substitutes "plant manager" for "Site Vice President." This changes the CTS by using less specific designations for the positions with the respective responsibilities.

These changes are acceptable because the responsibilities remain the same, but allow other documents to identify the specific company titles associated with the generic titles. This change is designated less restrictive because specific company titles associated responsibilities are deleted from the Technical Specifications.

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

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

   This change replaces specific company titles for specified responsibilities with less specific designations for these positions. Titles for positions of responsibility are not assumed to be initiators of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The company titles for specific plant responsibilities do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change replaces specific company titles for specified responsibilities with less specific designations for these positions. The ITS requirements are considered adequate because the responsibilities still have to be met by specific individuals, but the company titles of the individuals are not specified in the Technical Specifications. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.7

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.7 CTS 6.2.1.d states, "The management position responsible for training of the operating staff and the management position responsible for the quality assurance functions..." CTS 6.2.1.e states, "The management position responsible for health physics..." ITS 5.2.1.d states, "The individuals who train the operating staff, carry out health physics, or perform quality assurance functions..." This changes the CTS by using less specific designations for the positions with the respective responsibilities for the same functions.

These changes are acceptable because the responsibilities remain the same, but allow other documents to identify the specific company titles associated with the generic titles. This change is designated less restrictive because specific company titles associated responsibilities are deleted from the Technical Specifications.

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

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

This change replaces specific company titles for specified responsibilities with less specific designations for these positions. Titles for positions of responsibility are not assumed to be initiators of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The company titles for specific plant responsibilities do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change replaces specific company titles for specified responsibilities with less specific designations for these positions. The ITS requirements are considered adequate because the responsibilities still have to be met by specific individuals, but the company titles of the individuals are not specified in the Technical Specifications. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.8

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.8 CTS Table 6.2-1 includes requirements on SS, SRO, RO, AO, and STA position manning for each unit that are beyond what is required by 10 CFR 50.54(m)(2)(i). The ITS does not include these conditions. This changes the CTS by deleting certain criteria regarding how manning is distributed.

The intent of the conditions placed on unit staff manning is to state management policies regarding how the required positions are distributed between the two units at the site. This change is acceptable because this distribution can still be retained in accordance with management policy, but does not need to be retained in the ITS. The 10 CFR 50.54(m)(2)(i) requirements for staff manning are still required to be met. This change is designated less restrictive because conditions regarding the required staff manning are being deleted.

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

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

   This change deletes certain criteria for how shift crew composition is distributed. Shift crew composition distribution is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Shift manning requirements beyond the minimum required does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change deletes certain criteria for how shift crew composition is distributed. The ITS requirements are considered adequate for shift manning, and requirements for the exact distribution of the shift crew is not required to assure adequate manning. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.9

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.9 CTS Table 6.2-1 requires with either or both units in MODE 1, 2, 3 or 4, that four Auxiliary Operators (AOs) be part of the staff manning, two AOs assigned to each unit. CTS Table 6.2-1 also requires that with both units in MODE 5 or 6 or defueled, two Auxiliary Operators (AOs) be part of the staff manning, one AO assigned to each unit. ITS 5.2.2.a states, “A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each reactor which is operating in MODES 1, 2, 3, or 4." Therefore, with one unit in MODES 1, 2, 3, or 4, and one unit in MODE 5 or 6, the ITS would require 3 AOs while the CTS would require 4 AOs. This changes the CTS by only requiring one AO for each unit containing fuel, and an additional AO for each unit in MODES 1, 2, 3, or 4.

The purpose of the AO requirements in CTS Table 6.2-1 it to provide assurance that sufficient AOs are on the shift crew. This change is acceptable because it still provides a minimum number of AOs for the units in all the unit MODES defined in the ITS, which excludes a defueled reactor. It also makes the AO requirement more unit specific, designating the number of AOs required based on unit condition, and not requiring a second AO for a shutdown unit because the opposite unit is in MODES 1, 2, 3, or 4. This change is designated less restrictive because unit AO manning is reduced based on not being dependent on opposite unit MODE status, and no AOs being required for a defueled unit.

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

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

   This change deletes criteria for AO manning associated with the opposite unit. Shift crew composition for the opposite unit is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. AO manning criteria do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change deletes criteria for AO manning associated with the opposite unit. The ITS requirements are considered adequate for shift manning, and adequate for shift manning is not dependent on manning for the opposite unit. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.10

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.10 CTS Table 6.2-1, with regard to work hour procedures, states, "In addition, procedures will provide for documentation of authorized deviations from these guidelines and that the documentation is available for NRC review." ITS 5.0 does not include such a requirement. This changes the CTS by deleting a requirement to have a procedure for documentation of authorized deviations from the work hour guidelines and to have the documentation available for NRC review.

The purpose of the CTS Table 6.2-1 requirements regarding a procedure for documentation of authorized deviations from the work hour guidelines and to have the documentation available for NRC review is to assist in documenting and correcting guideline deviations. This change is acceptable because the work hour guidelines are still required to be met, but retaining the requirements for procedural documentation of authorized deviations will be retained under licensee control. This change is designated less restrictive because a requirement for procedural controls is being deleted from the CTS.

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

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

   This change deletes a requirement to have a procedure for documentation of authorized deviations from work hour guidelines and to have the documentation available for NRC review. These procedures and documentation are not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Maintenance of documentation regarding authorized deviations from work hour criteria does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change deletes a requirement to have a procedure for documentation of authorized deviations from work hour guidelines and to have the documentation available for NRC review. The ITS requirements are considered adequate to assure that work hour guidelines are met, and the procedure for documentation and availability of documentation is not essential in order to meet the guidelines. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.11

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.11 CTS 6.2.2.c references requirements for a health physics technician. CTS 6.12.1, footnote "*" describes a Health Physics technician allowance. CTS 6.12.2 references a responsibility of the Shift Supervisor on duty and/or the Plant Health Physicist. ITS 5.2.2.d references a radiation protection technician, and ITS 5.7.1 references Radiation Protection personnel, and ITS 5.7.2 references the radiation protection shift supervisor, radiation protection manager or his or her designee responsibilities, respectively. This changes the CTS by changing the titles of the personnel in the specified positions to more generic titles.

The purpose of these CTS requirements is to specify qualifications of people assigned particular duties. This change is acceptable because though the titles are more generic, as described in ANSI/ANS 3.1, the personnel will continue to be qualified to plant standards. These changes are designated as less restrictive because the titles of personnel in designated positions are less specific.

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

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

   This change replaces specific company titles for specified responsibilities with less specific designations for these positions. Titles for positions of responsibility are not assumed to be initiators of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The titles for specific plant responsibilities do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change replaces specific company titles for specified responsibilities with less specific designations for these positions. The ITS requirements are considered adequate because the responsibilities still have to be met by specific individuals, but the company titles of the individuals are not specified in the Technical Specifications. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.12

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.12 CTS 6.8.4 states that one of the programs to be established, implemented, and maintained is, "A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels." ITS 5.5.2 requires that the program minimize the same leakage. This changes the CTS by requiring the program provide controls to minimize instead of reduce leakage.

The purpose of the CTS 6.8.4 program for leakage from primary coolant sources outside containment is to provide guidance for the program which would reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This change is acceptable because the program will still keep the potential leakage to as low as practical levels, but will require leakage be minimized rather than reduced, which is more commensurate with the term as low as practical. This change is designated as less restrictive because a less stringent requirement is being applied to a program.

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

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

This change requires a program minimize rather than reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. Leakage from systems outside containment is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The program to minimize this leakage does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change requires a program minimize rather than reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The ITS requirements are considered adequate for controlling leakage by minimizing the leakage. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.13

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.13 ITS 5.7.2.f states, "Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device." CTS 6.12.2 does not include such an allowance. This changes the CTS by providing an additional method by which to control a high radiation area meeting the criteria of 5.7.2.

The purpose of ITS 5.7.2.f is to provide an adequate optional means of controlling access to a high radiation area described in ITS 5.7.2. This change is acceptable because it provides adequate controls for the areas addressed by ITS 5.7.2 without requiring controls for a much larger area that would restrict work and access while providing no substantial improvement in control of the area. This change is designated as less restrictive because it provides an additional option for controlling the areas addressed by ITS 5.7.2.

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

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

This change allows an additional option for how to control access to high radiation areas. Control of access to high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Requirements for access to high radiation areas do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows an additional option for how to control access to high radiation areas. The ITS requirements are considered adequate to provide assurance of adequate control of access to high radiation areas because the new alternative provided for access control provides clear and conspicuous indications and direction. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.15

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.15 CTS 6.9.1.4 requires annual reports described in CTS 6.9.1.5 be submitted prior to March 1 of each year. ITS 5.6.1 requires the Occupational Radiation Exposure Report to be submitted by April 30 of each year. This changes the CTS by allowing an additional 2 months to submit the Occupational Radiation Exposure Report each year.

The purpose of the due date for submitting the Occupational Radiation Exposure Report is to ensure it is provided in a reasonable period of time to the NRC for review. This change is acceptable because the report is still required to be submitted in a reasonable time frame. The change makes the due date consistent with the due dates for ITS 5.6.2 (Annual Radiological Environmental Operating Report) and ITS 5.6.3 (Radioactive Effluent Release Report). This change is designated as less restrictive because it allows more time to prepare and submit an annual report to the NRC.

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

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

   This change allows an additional two months to submit the Occupational Radiation Exposure Report. Submittal of the Occupational Radiation Exposure Report is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Submittal dates for reports do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change allows an additional two months to submit the Occupational Radiation Exposure Report. The ITS requirements are considered adequate for submittal of the Occupational Radiation Exposure Report in order to document and provide for review the information collected. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.16

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.16 CTS 6.12.1 states for high radiation areas, "...entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit." ITS 5.7.1.b and ITS 5.7.2.b state for high radiation areas, "Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures." This changes the CTS by allowing an equivalent document to be used for access control. The addition of details required in the RWP is addressed by DOC M.4.

The purpose of the specified phrase in CTS 6.12.1 is to designate the document through which access is controlled to the specified high radiation areas. This change is acceptable because a proper document is still required, but it may serve the same purpose as an RWP without having to be specifically an RWP. This change is designated a less restrictive because an alternate document may be used for access control in lieu of an RWP.

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

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

This change allows the requirements for access to high radiation areas in which the intensity of radiation is greater than 1000 mrem/hr to also apply to areas with ≥ 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. Requirements for access to high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Requirements for access to high radiation areas do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows the requirements for access to high radiation areas in which the intensity of radiation is greater than 1000 mrem/hr to also apply to areas with ≥ 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. The ITS requirements are considered adequate for control of access to high radiation areas, and this provides new guidance for access to the additional areas specified. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.17

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.17 Unit 1 CTS 6.12, High Radiation Area, footnote "*," states, "Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas." Unit 2 CTS 6.12, High Radiation Area, footnote "*," states, "Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas." ITS 5.7.1 states, "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." This changes the CTS by allowing Unit 1 personnel other than the Health Physics personnel to be exempt from the RWP issuance requirement, and for both Unit 1 and Unit 2, the personnel are allowed to use the exemption for the performance of assigned duties, not only radiation protection duties. These criteria apply in high radiation areas with exposure rates <1000 mrem/hr. Changing the term "Health Physics" to "radiation protection" is addressed by DOC L.11.

The purpose of CTS 6.12 footnote "*" is to provide an allowance for qualified personnel to not have to issue an RWP during the performance of their assigned radiation protection duties. This is because their training and the use of approved radiation protection procedures provides assurance that their personnel exposure will be within established limits. This change is acceptable because the escort of people by these trained individuals for any duties, not just for radiation protection, using approved radiation protection procedures, also provides assurance that the personnel exposure of the of the people being escorted will be within established limits. These changes are designated as less restrictive because a larger group of individuals will be eligible to be exempt from RWP issuance, and for a wider variety of duties.

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

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

   This change allows personnel other than Health Physics personnel to be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas. Specification of which personnel are exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Requirements for access to high radiation areas do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change allows personnel other than Health Physics personnel to be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas. The ITS requirements are considered adequate for high radiation area access control because stringent criteria are still being applied to all the personnel allowed access, similar to criteria applied previously. As a result, the change does not significantly reduce the margin of safety.

FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.18

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.18 CTS Table 6.2-1 states the qualifications for the person that assumes the control room command function during the absence of the Shift Supervisor, and excludes the STA as a person who can assume that function. ITS 5.1.2 does not include this exclusion of the STA. This changes the CTS by allowing an STA that holds a valid SRO license to assume the control room command function during the absence of the Shift Supervisor.

The purpose of the exclusion of the STA from being allowed to assume the control room command function during the absence of the Shift Supervisor is to provide assurance that the independent STA function is retained. This change is acceptable because the STA assuming the control room command function is a temporary state, and if the STA is qualified to assume the function, the STA is trained to man that position. There is no such explicit exclusion regarding manning in 10 CFR 50.54(m)(2)(i). This change is designated as a less restrictive change because an additional member of the shift crew composition is allowed to assume the control room command function during the absence of the Shift Supervisor.

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

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

   This change allows the STA to assume the control room command function if that person meets all the appropriate qualifications. Designation of the person with the control room command function is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Identifying which qualified personnel may assume the control room command function does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows the STA to assume the control room command function if that person meets all the appropriate qualifications. The ITS requirements are considered adequate for control room command function manning because the qualifications for the person assuming that function are not changing. The only change is expanding the group of people eligible to assume the function to the STA. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.19

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.19 CTS 6.4.1 states, "The Manager – Nuclear Training is responsible for ensuring that retraining and replacement training programs for the licensed facility staff meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4). Also, a retraining and replacement training program for non-licensed facility staff shall meet or exceed the recommendations of Section 5 of ANS 3.1 (12/79 Draft)*." CTS 6.4.1 footnote "*" states, "Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase."" ITS 5.0 does not include these requirements. This changes the CTS by not specifying who is responsible for ensuring the requirements of 10 CFR 55.59(c) and 55.31(a)(4) are met, and not specifying requirements for non-licensed facility staff training.

The purpose of CTS 6.4.1 is to assign responsibility for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and also to specify criteria for the training of the non-licensed facility staff. This change is acceptable because the requirements of 10 CFR 55.59(c) and 55.31(a)(4) are still required to be met, and the training of non-licensed facility staff is still expected to meet appropriate standards. Identification of the person responsible for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and specification of the training requirements to be met for non-licensed facility staff is addressed by plant processes. This change is designated as less restrictive because designation of the responsibility for meeting the requirements of 10 CFR 55.59(c) and 55.31(a)(4), and requirements for the training program for non-licensed facility staff are deleted.

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

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

   This change deletes the specification of the person responsible for ensuring the requirements of 10 CFR 55.59(c) and 55.31(a)(4) for training are met, and deletes the specification requirements for non-licensed facility staff training. Designation of responsibility for meeting regulatory requirements for training and specific non-licensed facility staff training requirements are not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Specifying the person responsible for ensuring training requirements are met and specifying requirements for non-licensed facility staff training does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change deletes the specification of the person responsible for ensuring the requirements of 10 CFR 55.59(c) and 55.31(a)(4) for training are met, and deletes the specification requirements for non-licensed facility staff training. The ITS requirements are considered adequate for meeting regulatory requirements and providing adequate non-licensed facility staff training because the facility is still responsible for meeting the regulatory requirements, and adequacy of non-licensed facility staff training is a licensee responsibility. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.21

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.21 ITS 5.6.1 allows dose assignments to various duty functions to be estimated using, among other things, an electronic dosimeter. CTS 6.9.1.5 does not include this allowance. This changes the CTS by including an electronic dosimeter as one of the ways by which dose assignments to various duty functions may be estimated.

The allowance in ITS 5.6.1 to use an electronic dosimeter as one of the ways by which dose assignments to various duty functions may be estimated allows the use of equipment that can provide valid measurements for this requirement. This change is acceptable because measurements from the electronic dosimeters will be used in conjunction with other equipment to make a best estimate. This change is designated less restrictive because it allows the use of equipment not previously allowed for performing a required estimate.

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

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

This change allows the option of using an electronic dosimeter for estimating dose assignments. Estimation of dose assignments is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Specifying which instruments may be used for estimating dose assignments does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows the option of using an electronic dosimeter for estimating dose assignments. The ITS requirements are considered adequate for dose assignment estimation because an electronic dosimeter is considered an adequate measurement device for dose assignment. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.22

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.22 Unit 1 CTS 4.4.5 Table 4.4-1 states that if an additional steam generator is in category C-3, one Action Required is, “Report to NRC & obtain approval prior to operation.” ITS Table 5.5.8-2 for the same condition states, “Report to NRC pursuant to 5.6.7.c.” This changes the CTS by not requiring obtaining NRC approval prior to operation in the event an additional steam generator is found to be in the category C-3.

The purpose of the reporting requirements of CTS 4.4.5 is to ensure the NRC is informed of steam generator tube inspection results which fall into the category C-3. CTS 4.4.5.5 and ITS 5.6.7 require results of steam generator tube inspections which fall into Category C-3 require prompt notification of the NRC pursuant to 10 CFR 50.72 and 10 CFR 50.73. This change is acceptable because the regulations provide adequate reporting requirements. Prompt reporting to the NRC is still required in accordance with regulations. The NRC may still require the licensee to obtain approval prior to unit operation, but the requirement will not be specified in the Technical Specifications. Unit 2 does not require the licensee to obtain approval, only appropriate reporting. This change is designated as less restrictive because it does not require input from the NRC before resuming unit operation.

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

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

   This change allows unit operation without obtaining NRC approval in the event an additional steam generator is found to be in the category C-3. Obtaining NRC approval for unit operation due to a particular technical condition is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Obtaining NRC approval for unit operation due to a particular technical condition does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change allows unit operation without obtaining NRC approval in the event an additional steam generator is found to be in the category C-3. The ITS requirements are considered adequate for steam generator Operability. Thus, NRC approval for unit operation is unnecessary since the appropriate technical requirements for Operability and unit operation are in the ITS. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.23

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.23 CTS 6.12.2 states, regarding areas in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates, "In addition, locked doors shall be provided to prevent unauthorized entry into such areas..." ITS 5.7.2 states, "...areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry." This changes the CTS by allowing the doors to be guarded as an option to locking them.

The purpose of CTS 6.12.2 is to state the means by which to prevent unauthorized access. This change is acceptable because adequate controls are maintained to prevent unauthorized access, while allowing reasonable flexibility regarding how to establish those controls. These changes are designated as less restrictive because it allows an additional means of preventing unauthorized entry into the specified high radiation area.

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

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

   This change allows areas with radiation levels ≥ 1000 mrem/hr to be provided with either locked or continuously guarded doors to prevent unauthorized entry, rather than requiring the doors be locked. Control of access to high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Requirements for access to high radiation areas do not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change allows areas with radiation levels ≥ 1000 mrem/hr to be provided with either locked or continuously guarded doors to prevent unauthorized entry, rather than requiring the doors be locked. The ITS requirements are considered adequate for radiation exposure control because either means adequately prevent unauthorized entry. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.24

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.24 CTS Table 6.2-1 states, "Procedures will be established to insure that NRC policy statement guidelines regarding work hours established for employees are followed." ITS 5.2.2.d states, "Administrative procedures shall be developed and implemented to limit working hours of personnel who perform safety related functions…" This changes the CTS by not referencing the NRC policy statement guidelines regarding work hours as the source of guidance for limiting work hours.

The purpose of the work hour control note in CTS Table 6.2-1 is to provide reasonable assurance that impaired performance caused by excessive work hours will not jeopardize safe plant operation. This change is acceptable because specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision making process to minimize the potential for impaired personnel performance, and that established procedure control processes will provide sufficient controls for changes to that procedure. Referencing the NRC policy statement guidelines regarding work hours is not required to accomplish this. This change is designated as a less restrictive change because the NRC policy statement guidelines regarding work hours is not specifically referenced in the Technical Specifications.

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

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

This change removes the reference to the NRC policy statement guidelines regarding work hours as the source of guidance for limiting work hours. Guidelines for work hours are not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The method for controlling work hours does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change removes the reference to the NRC policy statement guidelines regarding work hours as the source of guidance for limiting work hours. The ITS requirements are considered adequate for control of work hours because the ITS still requires controls include guidelines on working hours that ensure adequate shift coverage be maintained without routine heavy use of overtime. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.27

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.27 ITS 5.7.1.4.d.3 states that one of the options for devices an individual or group shall possess for radiation monitoring when entering a high radiation area with a dose rate not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation is, "A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area." ITS 5.7.2.4.d.2 states that one of the options for devices an individual or group shall possess when entering a high radiation area with a dose rate exceeding 1.0 rem/hour at 30 Centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or any surface penetrated by the radiation is, "A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area." CTS 6.12.1

and 6.12.2 do not contain these options for an individual or group. This changes the CTS by providing an additional device an individual entering these high radiation areas must possess for radiation monitoring.

The purpose of ITS 5.7.1.4.d.3 and ITS 5.7.2.4.d.2 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose of personnel in high radiation areas has been provided.

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

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

   This change provides an additional option for what an individual entering the specified high radiation areas must possess for radiation monitoring. Radiation monitoring for personnel entering these high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The method of radiation monitoring in high radiation areas does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change provides an additional option for what an individual entering the specified high radiation areas must possess for radiation monitoring. The ITS requirements are considered adequate for radiation monitoring for personnel entering high radiation areas because using a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote monitored receiver as described in the ITS is an adequate means of radiation monitoring. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.28

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.28 CTS 6.12.1.b states that one of the optional criteria that allow entry into a high radiation area is, "An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit." ITS 5.7.1.d.4 states, "A self reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and, (i) be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or (ii) be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance." ITS 5.7.2.d.3 reads the same as ITS 5.7.1.d.4, except the last phrase, "communicate with individuals in the area who are covered by such surveillance," is replaced with the phrase, "communicate with and control every individual in the area." This changes the CTS by deleting the discussion of positive controls over activities and performing radiation surveillances with a requirement for the monitoring device to have continuous dose rate displays and the responsibility to control dose rates in the area, and an option to perform the monitoring of personnel remotely using the specified equipment and processes.

The purpose of 6.12.1.c is to provide the option of monitoring the exposure of individuals in high radiation areas by a separate individual qualified in radiation procedures. This change is acceptable because it provides adequate means of monitoring the personnel in the high radiation areas, but provides added flexibility for how to do it. This change is designated as less restrictive because additional methods for monitoring personnel exposure are provided.

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

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

   This change replaces the discussion of positive controls over activities and performing radiation surveillances in the specified high radiation areas with a requirement for the required monitoring device to have continuous dose rate displays and the responsibility for the individual qualified in radiation protection procedures to control dose rates in the area. In addition, an option is added to perform the monitoring of personnel remotely using the specified equipment and processes. Radiation monitoring of personnel in high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The method of radiation monitoring in high radiation areas does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change replaces the discussion of positive controls over activities and performing radiation surveillances in the specified high radiation areas with a requirement for the required monitoring device to have continuous dose rate displays and the responsibility for the individual qualified in radiation protection procedures to control dose rates in the area. In addition, an option is added to perform the monitoring of personnel remotely using the specified equipment and processes. The ITS requirements are considered adequate for radiation monitoring for personnel entering high radiation areas because responsibility for monitoring radiation exposure is retained, and adequate alternate means of radiation monitoring are provided. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.29

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.29 ITS 5.7.2.4.d.4 states that one of the options for devices that an individual or group shall possess when entering a high radiation area with a dose rate exceeding 1.0 rem/hour at 30 Centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or any surface penetrated by the radiation is, "In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area." CTS 6.12.1 and 6.12.2 do not contain these options for an individual or group. This changes the CTS by providing an additional option for devices an individual entering these high radiation areas must possess.

The purpose of ITS 5.7.2.4.d.4 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide reliable means of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose of personnel in high radiation areas has been provided.

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

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

   This change allows an additional option for what an individual entering these high radiation areas must possess. Radiation monitoring of personnel in high radiation areas is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The method of radiation monitoring in high radiation areas does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change allows an additional option for what an individual entering these high radiation areas must possess. The ITS requirements are considered adequate for radiation monitoring for personnel entering high radiation areas because the new alternative provided for radiation monitoring also provides adequate indication of radiation dose rates in the area. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.30

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.30 CTS 6.8.2 states, "Each new procedure of 6.8.1 above, except 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved by the SNSOC prior to implementation as set forth in administrative procedures. Procedures of 6.8.1.d, 6.8.1.e, and 6.8.1.f shall be reviewed and approved as set forth in the facility's Security Plan, Emergency Plan, and section 6.5.1.6.m of the Technical Specifications, respectively." VTS 6.8.1.d is Security Program implementation. CTS 6.8.1.e is Emergency Plan implementation. CTS 6.8.1.f is Fire Protection Program Implementation. CTS 6.8.3 states, "Procedure changes that require a safety evaluation shall also be reviewed and approved by SNSOC. All other changes shall be independently reviewed and approved as programmatically discussed in the Updated Final Safety Analysis Report." ITS 5.0 does not include statements like those in CTS 6.8.2 and 6.8.3 regarding review and approval of procedures of CTS 6.8.1.d, 6.8.1.e, 6.8.1.f, and review and approval of changes as described in the UFSAR. This changes the CTS by not specifying how these procedures are reviewed and approved.

The purpose of the portions of CTS 6.8.2 and 6.8.3 of concern is to provide assurance that the referenced procedures are changed in accordance with the specified documents. This change is acceptable because the LCO requirements continue to ensure that the appropriate programs are maintained consistent with the licensing basis. The change deletes the requirement that changes to the specified procedures be reviewed and approved as stated in the referenced documents. Procedure changes are conducted in accordance with plant procedures. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

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

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

   This change deletes descriptions of how specified procedures are reviewed and approved. Review and approval of procedures is not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. The method of reviewing and approving procedures does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change deletes descriptions of how specified procedures are reviewed and approved. The ITS requirements are considered to provide adequate control of procedures, and not require direction for the review and approval of the specified procedures. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.31

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.31 CTS 6.8.4.e.5 states that the radioactive effluent control program shall include "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." ITS 5.5.4.e states that the radioactive effluent control program shall include "Determination of cumulative 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology and parameters in the ODCM at least every 31 days." This changes the CTS by not requiring that a projection of the dose contribution for the current calendar quarter and the current calendar year be performed every 31 days.

The purpose of the portions of CTS 6.8.4.e.5 is to determine the cumulative dose contributions for the current calendar quarter and current calendar year and to then project the dose contributions in the future. This is necessary to assess current and future compliance with offsite dose limits. This change is acceptable because the requirements continue to ensure that the appropriate programs are maintained consistent with the licensing basis. The current wording could be construed to require projection for the current quarter and current year. This misleading wording was promulgated in Generic Letter 89-01. The NRC has agreed that the proposed wording represents the intent of the requirements in their approval of TSTF-308, Revision 1. This change is designated as less restrictive because less stringent requirements are being applied in the ITS than were applied in the CTS.

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

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

This change eliminates the requirement to project dose contributions for radioactive effluents for the current calendar quarter and the current calendar year. Projection of dose contributions is not an initiator of any previously analyzed accident. Therefore, the change does not increase the probability of such accidents. Projection of dose contributions does not affect the ability of the plant to mitigate the consequences of previously analyzed accidents. As a result, the change does not significantly increase the consequences of an accident previously analyzed.

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

This change eliminates the requirement to project dose contributions for radioactive effluents for the current calendar quarter and the current calendar year. The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change eliminates the requirement to project dose contributions for radioactive effluents for the current calendar quarter and the current calendar year. The ITS requirements are considered to provide adequate monitoring of dose contributions from radioactive effluents. As a result, the change does not significantly reduce the margin of safety.