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ITS CHAPTER 5.0

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Department Leader, Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Department Leader, Operations 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 Control Room Supervisor (CRS) shall be responsible for the control room command function. During any absence of the CRS 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 CRS 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 shall be documented in the UFSAR;

b. The Vice President, Nuclear Production shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. The Senior Vice President, Nuclear 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.

5.2.2 Unit Staff

The unit staff organization shall include the following:

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

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

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

b. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

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 unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, 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 working hour guidelines shall be authorized in advance by personnel at the Director level or designees, in accordance with approved administrative procedures and with documentation of the basis for granting the deviation.

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

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

e. The Operations Department Leader or Operations Supervisor shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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 Regulatory Guide 1.8, September 1975 and ANSI/ANS 3.1-1978, except the Director, Site Radiation Protection shall meet or exceed the qualification of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

5.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operation (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).

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 to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

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),

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 Director, Site Chemistry; 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

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

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

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 portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sampling return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. 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.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

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 and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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;

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

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

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.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR.

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

(continued)

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

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 0, October 1971.

5.5.8 Inservice Testing Program

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

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

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

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

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

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the Inservice Inspection of steam generator tubes to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

5.5.9.1 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 5.5.9-1.

5.5.9.2 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 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. 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 (greater than 20%).

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

3. A tube inspection (pursuant to Specification 5.5.9.4a.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 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2 Steam Generator Tube Sample Selection and Inspection (continued)

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
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.

5.5.9.3 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

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

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 5.5.9-2 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 5.5.9.3a; 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 5.5.9-2 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 3.4.14.

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

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

4. A main steam line or feedwater line break.

5.5.9.4 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.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

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 or equal to 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 5.5.9.3c., above.

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

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 was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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 5.5.9-2.

(continued)

TABLE 5.5.9-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*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.

TABLE 5.5.9-2
STEAM GENERATORS 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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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.

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

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:

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.

5.5.11 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 the frequencies specified in Regulatory Guide 1.52, Revision 2, and in accordance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and AG-1 at the system flowrate specified below ± 10%.

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 Guide 1.52, Revision 2, and ANSI N510-1980, at the system flowrate specified as follows ± 10%:

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM
Hydrogen Purge Cleanup System (HPCS)	50 CFM

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 Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREFS	28,600 CFM
ESF PREACS	6,000 CFM
HPCS	50 CFM

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 Guide 1.52, Revision 2, and ANSI N510-1980 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 at a temperature of 80°C ± 0.5°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤ 1.0%	70%
ESF PREACS	≤ 1.0%	70%
HPCS	≤ 1.0%	70%

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREFS	8.4 inches water gauge	28,600 CFM
ESF PREACS	8.4 inches water gauge	6,000 CFM
HPCS	2.26 inches water gauge	50 CFM

e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage
ESF PREACS	> 19 kW
HPCS	> 0.5 kW

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks 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".

(continued)

5.5 Programs and Manuals (continued)

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

The program shall include:

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

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

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

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

5.5.13 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 as referenced in the UFSAR. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

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

3. A clear and bright appearance with proper color;

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

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

5.5.14 Technical Specifications (TS) Bases Control Program

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

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

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

A change in the TS incorporated in the license; or

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

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

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b 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).

5.5.15 Safety Functions Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

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

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

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Functions Determination Program (continued)

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

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

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

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

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

Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

    1. Overall air lock leakage rate is $\le 0.05\ L_a$ when tested at $\ge P_a$.

    2. For each door, leakage rate is $\le 0.01\ L_a$ when pressurized to ≥ 14.5 psig.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

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

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) 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, 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.

5.6.2 Annual Radiological Environmental Operating Report

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

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

(continued)

5.6 Reporting Requirements (continued)

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 in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

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

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

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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the shutdown cooling system suction line 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.

(continued)

5.6 Reporting Requirements (continued)

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. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Length CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

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:

1. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80" Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

7. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

8. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

9. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

10. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

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

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

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "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.

(continued)

5.6 Reporting Requirements (continued)

5.6.7 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.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

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

c. Identification of tubes plugged.

Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation 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

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 Exposure Permit (REP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technicians) or personnel continuously escorted by such individuals may be exempt from the REP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

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

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

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

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

(continued)

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem 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 Control Room Supervisor on duty or Radiation Protection supervision. Doors shall remain locked, except during periods of access by personnel under an approved REP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the REP, 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 accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose in excess of 1000 mrem (measurement made at 30 cm from source of radioactivity), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, 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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<6.0> 5.0 ADMINISTRATIVE CONTROLS

<6.1> 5.1 Responsibility

<6.1.1> 5.1.1 The ~~[Plant Superintendent]~~ Department Leader, Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. 2

<6.5.2.3> <6.5.2.5> 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. 2

<6.1.2> 5.1.2 The ~~[Shift Supervisor (SS)]~~ Control Room Supervisor (CRS) shall be responsible for the control room command function. During any absence of the [SS] CRS 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. 2

<Table 6.2-1> 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.

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<6.0> 5.0 ADMINISTRATIVE CONTROLS

<6.2> 5.2 Organization

<6.2.1> 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit 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]~~ {UFSAR} (2);

<6.2.1.c> b. The ~~[Plant Superintendent]~~ {Vice President, Nuclear Production} (2) 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;

<6.2.1.b> c. The ~~[a specified corporate executive position]~~ {Senior Vice President, Nuclear} (2) 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> 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:

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

(continued)

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<6.2> 5.2 Organization

Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). (11)

<6.2.2> 5.2.2 Unit Staff (continued)

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.~~ (2)

<6.2.2.b> ~~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.~~ (9)

<Table 6.2-1> ~~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 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.

<6.2.2.c> <P.6-1a asterisk Note> ~~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. (2)

<6.2.2.1.a> ~~e.~~ d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, ~~health physicists~~, radiation protection technicians, auxiliary operators, and key maintenance personnel). (2)

<6.2.2.1.b> ~~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:~~ (this excludes the STA and PVNGS Fire Department working hours) (10)

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

(continued)

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime. (10)

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<6.2> 5.2 Organization

<6.2.2> 5.2.2 Unit Staff (continued)

<6.2.2.1.b>

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

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

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

<6.2.2.1.c> Any deviation from the above working hours guidelines shall be authorized in advance by the [Plant Superintendent] or his designee personnel at the Director level, 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. S 2 10

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee these authorized individuals to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized. S 2

~~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).~~ 2

<6.3.1> e→f The [Operations Manager] or Assistant Operations [Manager] Supervisor shall hold an SRO license. Department Leader 2

<6.2.4.1> 5→g The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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<6.0> 5.0 ADMINISTRATIVE CONTROLS

<6.3> 5.3 Unit Staff Qualifications

~~Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ (2)

<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]. 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].~~ (2)

September 1975 and ANSI/ANS 3.1-1978, except the Director, Site Radiation Protection shall meet or exceed the qualification of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

5.3.2 For the purpose of 10 CFR 55.4, a licensed senior react operator (SRO) and a licensed reactor operator (RO) are individuals who, in addition to meeting the requirements c 5.3.1, perform the functions described in 10 CFR 50.54(m) (11)

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<6.0>
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>
a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

<6.8.1.a>
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];

<6.8.1.j>
c. Quality assurance for effluent and environmental monitoring;

<6.8.1.f>
d. Fire Protection Program implementation; and

<6.8.1.i>
e. All programs specified in Specification 5.5.

<6.8.1.m>
<6.8.1.g>
[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

<NOTE 2> Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.]

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<6.0> 5.0 ADMINISTRATIVE CONTROLS

<6.8> 5.5 Programs and Manuals

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

<6.14> 5.5.1 <u>Offsite Dose Calculation Manual (ODCM)</u>

<1.18>

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

<1.18>

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.6.2] and Specification [5.6.3].

<6.14> Licensee initiated changes to the ODCM:

<6.14.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),

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;

<6.14.b> b. Shall become effective after the approval of the [Plant Superintendent]; and {Director, Site Chemistry} 2

<6.14.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

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<5.8> 5.5 Programs and Manuals

<6.14> 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

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

<6.8.4.a> 5.5.2 Primary Coolant Sources Outside Containment

<4.5.2.e.4>

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:

INSERT 1

2

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.e> 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.g> 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

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recirculation portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sample return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem.

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5.5 Programs and Manuals

<6.8.4.g> 5.5.4 Radioactive Effluent Controls Program (continued)

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

<6.8.4.g(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; (12)

<6.8.4.g(2)> b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, 10 times the concentration values in conforming to ~~10 CFR 20~~, Appendix B, Table 2, Column 2; to 10 CFR 20.1001 - 20.2402

<6.8.4.g(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.g(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.g(5)> e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

<6.8.4.g(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;

from the site   at or   (12)

<6.8.4.g(7)> g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~;

INSERT 1

(continued)

INSERT FOR 5.5.4 ITEM g.

INSERT 1

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and 12

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

INSERT PAGE 5.0-9

<DOC>
<CTS>

<6.8> 5.5 Programs and Manuals

<6.8.4.g> 5.5.4 Radioactive Effluent Controls Program (continued)

<6.8.4.g(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;

<6.8.4.g(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.g(10)>

j. Limitations on the annual dose or dose commitment to any member of the public, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.
{beyond the site boundary} 12

<5.7.1> 5.5.5 Component Cyclic or Transient Limit {UFSAR} {3.9.1.1} 2

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

<4.6.1.6.2> 5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

<4.6.1.6.3>
<4.6.1.6.4>

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]~~. {as described in Section 1.8 of the UFSAR} 2

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

<4.4.9> 5.5.7 Reactor Coolant Pump Flywheel Inspection Program

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~~. {0, October 1971} 2

(continued)

<DOC>
<CTS>

<6.8> 5.5 Programs and Manuals (continued)

<4.0.5> 5.5.8 Inservice Testing Program

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

<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

<4.0.5.c> b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

<DOC A.16> c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

<4.0.5.e> d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

3.4.4 5.5.9 Steam Generator (SG) Tube Surveillance Program

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

INSERT 1

(continued)

This program provides controls for the Inservice Inspection of steam generator tubes to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

1

INSERT FOR 5.5.9

INSERT 1

REACTOR COOLANT SYSTEM

3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

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

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{cold} above 210°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

5.5.9.1 ~~4.4.4.1~~ 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~~. 5.5.9-1

5.5.9.2 ~~4.4.4.2~~ 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~~. The 5.5.9-2
5.5.9.3 inservice inspection of steam generator tubes shall be performed at the frequencies specified in ~~Specification 4.4.4.3~~ and the inspected tubes shall be verified acceptable per the acceptance criteria of ~~Specification 4.4.4.4~~. 5.5.9.4
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:

INSERT FOR 5.5.9

INSERT 1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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

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

5.5.9

3. A tube inspection (pursuant to Specification ~~4.4.4~~.4a.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 ~~4.4-2~~ 5.5.9-2) 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
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.

INSERT FOR 5.5.9

INSERT 1

5.5.9.3 ~~4.4.4.3~~ 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 calender 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 ~~4.4-2~~ 5.5.9-2 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 ~~4.4.4~~ 5.5.9.3a.; 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 ~~4.4-2~~ 5.5.9-2 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 ~~3.4.5.2~~ 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 main steam line or feedwater line break.

~~*Except that the inservice inspection due not later than July 1991 may be deferred until the end of fuel Cycle 3, but not beyond March 1992.~~

INSERT PAGE 5.0-11c

INSERT FOR 5.5.9

INSERT 1

5.5.9.4 ~~4.4.4.4~~ 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 or equal to 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 4.4.4 3c., above.

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

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

INSERT PAGE 5.0-11d

INSERT FOR 5.5.9

INSERT 1

SURVEILLANCE REQUIREMENTS (Continued)

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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~~. 5.5.9-2

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special 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 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

INSERT PAGE 5.0-11e

INSERT FOR 5.5.9

INSERT 1

TABLE ~~4.4-1~~ 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*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.

INSERT PAGE 5.0-145

Palo Verde - Units 1, 2, 3

3/4 4-17

INSERT PAGE 5.0-11g

TABLE ~~4.4-2~~ 5.5.9-2

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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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

INSERT FOR 5.5.9

INSERT 1

<DOC>
<CTS>

<6.8> 5.5 Programs and Manuals (continued)

<6.8.4.c> 5.5.10 Secondary Water Chemistry Program

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:

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.

<4.6.4.3> <4.7.7> <4.7.8> 5.5.11 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 the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ~~ASME~~ N510-~~1989~~, and AG-1] at the system flowrate specified below [± 10%].

1.52, Revision 2
ANSI
1980
2

<4.6.4.3.b.1, b.3, e>
<4.7.7.b.1, b.3, e>
<4.7.8.b.1, b.3, 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~~

≤1.0
ANSI
2

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

1980 N510-[1989], at the system flowrate specified as follows [± 10%]: (2)

ESF Ventilation System	Flowrate
[ ]	[ ]

INSERT 1 (2)

<4.6.4.3.f> <4.7.7.f> <4.7.8.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, and ASME ANSI N510-[1989] 1980 at the system flowrate specified as follows [± 10%]: (2)

ESF Ventilation System	Flowrate
[ ]	[ ]

INSERT 2 (2)

<4.6.4.3.b2,c> <4.7.7.c> <4.7.8.c> 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 Guide 1.52, Revision 2] and ANSI N510-1980, shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-[1989] 1979 at a temperature of < [30°C] 80°C ±0.5°C and greater than or equal to the relative humidity specified as follows: (2)

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

INSERT 3 (2)

(continued)

INSERT FOR 5.5.11
ITEMS a., b., AND c.

INSERT 1

Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered /Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM
Hydrogen Purge Cleanup System (HPCS)	50 CFM

INSERT 2

CREFS	28,600 CFM
ESF PREACS	6,000 CFM
HPCS	50 CFM

INSERT 3

CREFS	≤ 1.0%	70%
ESF PREACS	≤ 1.0%	70%
HPCS	≤ 1.0%	70%

INSERT PAGE 5.0-13

<DOC>
<CTS>

<6.8> 5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).~~ (1)

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

<4.6.4.3.d.1>
<4.7.7.d>
<4.7.8.d>

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-~~1989~~] at the system flowrate specified as follows [± 10%]: (2)

ANSI 1980

ESF Ventilation System	Delta P	Flowrate
INSERT 1 → [ ]	[ ]	[ ]

(2)

<4.6.4.3.d.2>
<DOC M.5>

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

ANSI 1980

ESF Ventilation System	Wattage
INSERT 2 → [ ]	[ ]

(2)

<DOC A.23> The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

<3.11.1>
<3.11.2>
<3.11.3>

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system,] and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The (2)

(continued)

INSERT FOR 5.5.11
ITEMS d. AND e.

INSERT 1

CREFS	8.4 inches water gauge	28,600 CFM
ESF PREACS	8.4 inches water gauge	6,000 CFM
HPCS	2.26 inches water gauge	50 CFM

INSERT 2

ESF PREACS	> 19 kW
HPCS	> 0.5 kW

INSERT PAGE 5.0-14

5.5 Programs and Manuals

<3.11.1>
<3.11.2>
<3.11.3>

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

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

The program shall include:

<3.11.2>

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

<3.11.3>

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 (2)

<3.11.1>

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

<DOC A.22>

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

(continued)

<DOC>
<CTS>
<6.8>

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program

<DOC M.4>

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: (3)

as referenced in the UFSAR

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

   3. A clear and bright appearance with proper color;

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

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3. 92 (4)

<DOC M4> 5.5.14 Technical Specifications (TS) Bases Control Program

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

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

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

   A change in the TS incorporated in the license; or

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

(continued)

<DOC>
<CTS>
<6.8> 5.5 Programs and Manuals

<DOC M.4> 5.5.14 Technical Specifications (TS) Bases Control Program (continued)

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

d. Proposed changes that meet the criteria of Specification 5.5.14b 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).

<DOC M.4> 5.5.15 Safety Functions Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

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

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

(continued)

<DOC>
<CTS>
<6.8>

5.5 Programs and Manuals

<DOC M.4> 5.5.15 Safety Functions Determination Program (continued)

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

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

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

<DOC>
<CTS>

5.5 Programs and Manuals

<6.16> 5.5.16 Containment Leakage Rate Testing Program 2

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995], as modified by the following exceptions:

[a/ /.....]

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is [45] psig]. The containment design pressure is [50] psig]. 52.0 60

The maximum allowable containment leakage rate, L_a, at P_a, shall be [1] % of containment air weight per day. 0.1

Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

   2) For each door, leakage rate is $\leq$ [0.01 L_a] when pressurized to $\geq$ [10] psig]. 14.5

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

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

<DOC>
<CTS>
<6.0>

5.0 ADMINISTRATIVE CONTROLS

<6.9>

5.6 Reporting Requirements

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

<6.9.1.5> 5.6.1 Occupational Radiation Exposure Report

<Page 6-19 Asterisk Note>

-----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) 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 initial criticality.] 2

<Page 6-19 Double Asterisk Note>

6.9.1.7 5.6.2 Annual Radiological Environmental Operating Report

<Page 6-20 Asterisk Note>

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

<6.9.1.7>

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

(continued)

<DOC>
<CTS>

<6.9> 5.6 Reporting Requirements

<6.9.1.7> 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.

<App. B, 5.4.1> 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.

<6.9.1.8> 5.6.3 Radioactive Effluent Release Report

<Page 6-20 asterisk Note>

-----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.

<6.9.1.8> 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 50, Appendix I, Section IV.B.1.

<6.9.6> 5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience*, including documentation of all challenges to the pressurizer 2

(continued)

<DOC>
<CTS>

<6.9> 5.6 Reporting Requirements

<6.9.6> 5.6.4 <u>Monthly Operating Reports</u> (continued) Shutdown cooling system suction line relief valves (2)

<3.4.8.3>

~~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.

<6.9.1.9> 5.6.5 <u>CORE OPERATING LIMITS REPORT (COLR)</u>

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.]~~ (2)

<6.9.1.10>

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.]~~ (2)

6.9.1.10 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.

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

~~5.6.6 <u>Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)</u>~~ (2)

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

(continued)

INSERT FOR 5.6.5

~~ADMINISTRATIVE CONTROLS~~

~~CORE OPERATING LIMITS REPORT~~

~~6.9.1.9 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:~~

INSERT 1

1 ~~a.~~ Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1~~.1~~

2 ~~b.~~ Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.~~1.~~2

3 ~~c.~~ Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.~~1.3~~ 4 System

4 ~~d.~~ Boron Dilution Alarms for Specification ~~3.1.2.7~~ 3.3.12

CEA Alignment ~~e.~~ ~~Movable Control Assemblies - CEA Position~~ for Specification 3.1.~~3.1~~ 5

5 ~~f.~~ Regulating CEA Insertion Limits for Specification 3.1.~~3.6~~ 7

6 ~~g.~~ Part Length CEA Insertion Limits for Specification 3.1.~~3.7~~ 8

7 ~~h.~~ Linear Heat Rate for Specification 3.2.1

8 ~~i.~~ Azimuthal Power Tilt - T_q for Specification 3.2.3

9 ~~j.~~ DNBR ~~Margin~~ for Specification 3.2.4

10 ~~k.~~ Axial Shape Index for Specification 3.2.~~7~~ 5

11 ~~l.~~ Boron Concentration (Mode 6) for Specification 3.9.1

12

~~6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:~~

INSERT 2

1 ~~a.~~ "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.~~3.6~~ 7, Regulating CEA Insertion Limits).

2 ~~b.~~ "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1~~.1~~, Shutdown Margin - Reactor Trip Breakers Open; 3.1.~~1.~~2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.~~1.3~~ 4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.~~3.6~~ 7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3 ~~c.~~ "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.~~1.~~2, Reactor Trip Breakers Closed; 3.1.~~1.3~~ 4, Moderator Temperature Coefficient BOL and EOL limits; ~~3.1.2.7~~ 3.3.12, Boron Dilution Alarms; 3.1.~~3.1~~ 5, ~~Movable Control Assemblies - CEA Position~~ CEA Alignment; 3.1.~~3.6~~ 7, Regulating CEA Insertion Limits; 3.1.~~3.7~~ 8, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q). Shutdown Margin System

4 ~~d.~~ "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR ~~Margin~~ and 3.2.~~7~~ 5 Axial Shape Index).

INSERT PAGE 5.0-21

INSERT FOR 5.6.5

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

INSERT 2 5-13. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

6-14. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

7-15. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

8-16. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f. 5.6.5.b.6

9-17. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g. 5.6.5.b.7

10-18. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11-19. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.s. 5.6.5.b.10

~~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 analysis limits) of the safety analysis are met.~~

~~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.~~

<DOC>
<CTS>
<6.9>

5.6 Reporting Requirements

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

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

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

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

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

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

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

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

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

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

5

(continued)

<DOC>
<CTS>

<6.9> 5.6 Reporting Requirements

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

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

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

5

5.6.7 EDG Failures Report

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

8

3.3.3.6 TABLE 3.3-10 Actions 31 and 32

5.6.8 7 PAM Report

When a report is required by Condition B or G of LCO 3.3.11 10, "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.

2

<4.6.1.6.5> 5.6.9 8 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

(continued)

<DOC>
<CTS>

<6.9> 5.6 Reporting Requirements

<4.6.1.6.5> 5.6.~~8~~ 8 [Tendon Surveillance Report (continued)

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

<4.4.4.5> 5.6.~~10~~ 9 Steam Generator Tube Inspector Report

INSERT 1

~~[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.]~~

~~[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.]~~ 2

INSERT FOR 5.6.10

SURVEILLANCE REQUIREMENTS (Continued)

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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.

4.4.4.5 Reports

INSERT 1

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special 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 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

INSERT PAGE 5.0-24

<POC>
<CTS>

<6.0> 5.0 ADMINISTRATIVE CONTROLS

<6.12> 5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [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.

≤ 6

REP

Exposure 2

Radiation Protection

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

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

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

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

2

REP

Section Leader or designated alternate

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

accessible to personnel 6

2

Control Room Supervisor

Radiation Protection

(continued)

such that an individual could receive in 1 hour a dose greater than

6

<6.12> *5.7 High Radiation Area*

5.7.2 (continued)

REP

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. (2)

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. (6)

such that an individual could receive in 1 hour a dose in excess of (6)

(measurement made at 30 cm. from source of radioactivity) (7)

NUREG-1432 EXCEPTIONS
CHAPTER 5.0

1. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

2. The plant specific titles, nomenclature, number, parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS.

3. CTS 3.8.1.3.1.2 identifies specific ASTM requirements. NUREG 5.5.13 has requirements that are "in accordance with applicable ASTM Standards." ITS 5.5.13 adds "as referenced in the UFSAR" to the NUREG statement. This change ensures that the location of the applicable ASTM Standards for PVNGS is clearly identified.

4. CTS 3.8.1.3.2 has a sample frequency of 92 days. NUREG 5.5.13 has a frequency of 31 days. ITS 5.5.13 is changed to retain the current licensing basis. Fuel oil degradation is normally a function of water content of the fuel oil. This is due to the climatic conditions at the plant. Due to the arid climate at PVNGS, the 92 day frequency provides sufficient time to detect fuel oil degradation. It should be noted that PVNGS has not experienced fuel oil degradation indicated by viscosity or sediment. Therefore, the current licensing basis has been determined to be appropriate for this surveillance requirement exceeding limits.

5. NUREG 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report, is not used in ITS. Therefore, this report is deleted from ITS.

6. TS 5.7.1 and 5.7.2 are changed to reflect the requirements in the latest version of 10 CFR 20 and the current licensing basis for PVNGS. These changes are consistent with the requirements of 10 CFR 20. Therefore, the current licensing basis has been determined to be appropriate.

7. CTS 6.12.2 Note ** states that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS 5.7.2 (NUREG exception) to ensure that it is clear where the dose is measured. ITS 5.7.2 provides requirements "In addition" to the requirements in ITS 5.7.1. ITS 5.7.1 references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20.

8. NUREG 1432 contains a requirement for an EDG failure report. This requirement has been deleted in the ITS in accordance with the guidance of Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994. PVNGS will implement a maintenance program for monitoring and maintaining diesel generator performance in accordance with the provisions of the maintenance rule. No additional requirements are needed to be included in ITS Chapter 5.0 for EDG Failures.

9. CTS 6.2.2.b and NUREG 5.2.2.b are deleted to incorporate the changes recommended by NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997). These paragraphs provided requirements that are redundant to 10 CFR 50.54(m)(2)(iii).

10. CTS 6.2.2.1.b, CTS 6.2.2.1.c, and NUREG 5.2.2.e provided specific working hour limits for plant staff. These requirements are revised to allow use of administrative procedures to control working hours. The change incorporates the recommendations of NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997).

11. STS 5.3.2 is added, and CTS Table 6.2-1 and NUREG 5.2.2.c are revised to ensure that there is no misunderstanding when complying with 10 CFR 55.4 requirements. These changes incorporate the recommendations of NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997).

12. CTS 6.8.4.g and NUREG 5.5.4 are revised to incorporate changes to 10 CFR 20 and 10 CFR 50.36a. These changes are intended to eliminate possible problems with implementation of the revised 10 CFR 20 requirements. These changes incorporate the recommendations of NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997).

PVNGS CTS
CHAPTER 5.0
MARK UP

CHAPTER 5.0
(1.0/5.0)

A.1

K_{N-1}

1.16 K_{N-1} is the k effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full-length CEA of the highest worth is fully withdrawn.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

ITS 1.0

ITS 5.0

5.5.1

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1a

~~1.18~~ 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 contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring 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 6.9.1.7 and 6.9.1.8.

5.5.1.b

5.0

ITS 1.0

OPERABLE - OPERABILITY

1.19 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, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

CHAPTER 5.0

(3.0/5.0)

APPLICABILITY

A.1

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 specified surveillance interval.

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

A.1

This program provides controls

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. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

ITS 3.0

ITS 5.0

5.5.8

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

including applicable supports

~~a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing 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).~~ LA.21

A.14

5.5.8a

~~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:~~ LA.21

Testing frequency A.1

A.1

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.a ~~4.0.5~~ (Continued)

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
Yearly or annually	At least once per 366 days
Bienially or every 2 years	At least once per 731 days

A.17

5.5.8.b ~~c.~~ The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities. LA.21

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements. A.15

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

INSERT 5.5.8.c A.16

CHAPTER 5.0
(3.4.14/5.0)

REACTOR COOLANT SYSTEM

3/4.4.4 STEAM GENERATORS

A.1

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

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

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{cold} above 210°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

ITS 3.4.14

ITS 5.0

5.5.9.1 ~~4.4.4.1~~ 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.

5.5.9.2 ~~4.4.4.2~~ 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. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. 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:

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the Inservice Inspection of steam generator tubes to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

A.1

CHAPTER 5.0

A.1

SURVEILLANCE REQUIREMENTS (Continued)

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

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

3. A tube inspection (pursuant to Specification 4.4.4.4a.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 4.4-2) 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
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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A.1

5.5.9.3 ~~4.4.4.3~~ 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 calender 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 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 4.4.4.3a.; 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 4.4-2 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 3.4.5.2.

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

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

   4. A main steam line or feedwater line break.

*Except that the inservice inspection due not later than July 1991 may be deferred until the end of fuel Cycle 3, but not beyond March 1992.

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4.1

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.4 ~~4.4.4.4~~ 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 or equal to 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 4.4.4 3c., above.

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

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

CHAPTER 5.0

A.1

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.4

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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.

5.6.8 ~~4.4.4.5~~ Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special 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 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

A.12

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	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*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.

A.1

CHAPTER 5.0

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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	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

A.1

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CHAPTER 5.0
(3.4.13/5.0)

A.1

3.4.13 LIMITING CONDITION FOR OPERATION (Continued)

5.0

5.6.4 ~~e.~~ In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, ~~a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.~~ A.18

ITS 5.0

ITS 3.4.13 f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise verify alignment every 12 hours.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

shall be included in the monthly operating report.

CHAPTER 5.0
(3.5.3/3.5.6/5.0)

EMERGENCY CORE COOLING SYSTEMS

A.1

SURVEILLANCE REQUIREMENTS (Continued)

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

ITS 3.5.3

ITS 3.5.6

2. Verifying that a minimum total of 464 cubic feet of solid granular anhydrous trisodium phosphate (TSP) is contained within the TSP storage baskets.

3. Verifying that when a representative sample of 0.055 ± 0.001 lb of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

ITS 3.5.6

ITS 3.5.3

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on (SIAS and RAS) test signal(s).

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a. High pressure safety injection pump.

b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI, LPSI and CS pump minimum bypass recirculation flow line isolation valves and combined SI mini-flow valve close, and the LPSI pumps stop.

ITS 3.5.3

ITS 5.0

5.5.2

~~4.~~ Conducting an inspection of all ECCS piping outside of containment, which is in contact with recirculation sump inventory during LOCA conditions and verifying that the total measured leakage from piping and components is less than 1 gpm when pressurized to at least 40 psig.

LA.22

ITS 5.0

ITS 3.5.3

f. By verifying that each of the following pumps develops the indicated differential pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1761 psid.

2. Low pressure safety injection pump greater than or equal to 165 psid.

CHAPTER 5.0
(3.6.2/5.0)

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

A.1

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

ITS 5.0

5.5.16 ~~b.~~ An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 49.5 psig. 52.0 A.25

ITS 5.0

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. 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, or

2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. The provisions of Specification 3.0.4 are not applicable.

b. With the 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

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage to be less than or equal to 0.01 L_a when determined with the volume between the door seals pressurized to greater than or equal to 14.5 $\pm$ 0.5 psig, for at least 15 minutes,

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

CHAPTER 5.0
(3.6.1/5.0)

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

A.1

ITS 3.6.1

ITS 5.0

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

a. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6 except for Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 15 days, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. LA.23

b. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 72 hours, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ITS 5.0

ITS 3.6.1

SURVEILLANCE REQUIREMENTS

ITS 3.6.1

ITS 5.0

4.6.1.6.1 The structural integrity of the containment vessel shall be demonstrated at the end of 1, 3 and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. All of the acceptance testing of tendon and visual examinations of end anchorages, adjacent concrete surfaces and containment vessel surfaces shall be performed sequentially and within the same time frame.

4.6.1.6.2 The structural integrity of the tendons shall be demonstrated by:

a. Determining from a random but representative sample of at least 10 tendons (6 hoop and 4 inverted U) that each group (hoop, and inverted U) has an observed lift-off force within the predicted limits for that group. For each subsequent inspection one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows: LA.24

INSERT 5.5.6

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies. A.19

CHAPTER 5.0

A.1

SURVEILLANCE REQUIREMENTS (Continued)

1) If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability;

2) If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence;

LA.24

3) If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be performed to determine the cause and extent of sucn occurrence;

4) If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as below the acceptance criteria for containment vessel structural integrity; and

5) Unless there is degradation of the containment vessel below the acceptance criteria during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 hoop and 3 inverted U).

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires. A previously stresses tendon wire or strands from one tendon of each group shall be removed for testing and examination over the entire length to determine (which should include the broken wire if so identified) that:

1) The tendon wires are free of corrosion, cracks, and damage;
2) There are no changes in the presence or physical appearance of the sheathing filler-grease; and

CHAPTER 5.0

A.1

SURVEILLANCE REQUIREMENTS (Continued)

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidenced that structural integrity is below the acceptance criteria.

c. Performing tendon retensioning of those tendons detensioned for inspection to at least force level recorded prior to detensioning or the predicted value, whichever is greater, with the tolerance within minus zero to plus 6%, except that the final seating force shall be such that the stress in the wire or strand shall not exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons, the stress in the tendon shall not exceed 80% of its ultimate strength, and the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during installation, an investigation shall be made to ensure that the difference is not related to wire failures or slips of wires in anchorages; and LA.24

d. Verifying the OPERABILITY of the sheathing filler-grease by assuring:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified as follows:

Water content	0 - 5% by wt.
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm
Sulfides	0 - 5 ppm
Reserved Alkalinity (Base Numbers)	0 - 50% of the installed value (installed value 0-5 for older grease).

4.6.1.6.3 As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. For those containments in multiple unit plants for which only visual inspection need be performed, tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load-bearing components of the anchorages. The surrounding concrete shall also be checked visually for indication of any abnormal condition.

CHAPTER 5.0

A.1

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.4 The exterior surface of the containment vessel shall be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage, each of which can be considered as evidence that the structural integrity is below the acceptance criteria.

4.6.1.6.5 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

LA.24

TABLE 4.6-1

TENDON SURVEILLANCE - FIRST YEAR

Tendon No.	Visual Inspection	Monitor Forces	Detension Tendon	Remove Wire	Test Wire
V32	X	X	No	No	No
V43	X	X	No	No	No
V62	X	X	X	X	X
V75*	X	X	A	A	A
H13-007*	X	X	X	X	X
H13-021	X	X	No	No	No
H21-037	X	X	No	No	No
H21-044	X	X	No	No	No
H32-016	X	X	No	No	No
H32-030	X	X	A	A	A

LA.74

Notes:

1. "X" means the tendon shown shall be inspected for the stated requirements during this surveillance.
2. "A" means the tendon shown shall be inspected for the stated requirements during the next or second surveillance.
3. "No" means that inspection is not required for that tendon.
4. "*" means control tendon.

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TABLE 4.6-2

TENDON LIFT-OFF FORCE FIRST YEAR
U-TENDONS

TENDON NUMBER	TENDON END	MAXIMUM (kips)	MINIMUM (kips)
V32	Shop	1463	1343
	Field	1510	1386
V43	Shop	1436	1364
	Field	1486	1364
V62	Shop	1475	1354
	Field	1486	1364
V75	Shop	1527	1402
	Field	1504	1380

HOOP TENDONS

TENDON NUMBER	TENDON END	MAXIMUM (kips)	MINIMUM (kips)
H13-007	Shop	1428	1300
	Field	1451	1321
H13-021	Shop	1515	1380
	Field	1491	1358
H21-037	Shop	1505	1371
	Field	1446	1317
H21-044	Shop	1484	1360
	Field	1530	1403
H32-016	Shop	1411	1282
	Field	1457	1324
H32-030	Shop	1473	1330
	Field	1473	1330

LA24

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(3.6.3/5.0)

A.1

SURVEILLANCE REQUIREMENTS (Continued)

ITS 3.6.3 4.6.3.3 The isolation time of each power operated or automatic valve used in CIAS, CPIAS, or CSAS shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

ITS 5.0

5.5.16 ~~4.6.3.4~~ The containment isolation check valves shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

5.5.8 ~~4.6.3.5~~ The containment isolation valves used as safety/relief, normally open—ESF actuated closed, or required open during accident conditions shall be demonstrated OPERABLE as required by ~~Specification 4.0.5~~ Inservice Testing Program and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured*** in their actuated position are considered operable pursuant to this specification.

5.5.8.d

ITS 5.0

ITS 3.6.3

A.1

4.6.3.6 The manual containment isolation valves (normally closed/post accident closed valves) shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

*Locked, sealed, or otherwise prevented from unintentional operation.

CHAPTER 5.0
(3.6.7/5.0)

CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

A.1

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

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:

3.6.7

ITS 5.0

5.5.11.a ~~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 50 scfm ± 10%.

5.5.11.c ~~2.~~ Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,** meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.**

LA.25

ITS 5.0

ITS 3.6.7 *With less than two hydrogen recombiners OPERABLE.

ITS 5.0 ~~**ANSI N509-1980 is applicable for this specification.~~

LA.29

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.11.a ~~3.~~ Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.11.c ~~c.~~ After ~~every 720 hours of~~ charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with ~~Regulatory Position C.6.b of~~ Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria ~~of Regulatory Position C.6.a~~ of Regulatory Guide 1.52, Revision 2, March 1978.* LA.25

5.5.11 ~~d.~~ ~~At least once per 18 months by:~~ As specified in Regulatory Guide 1.52

5.5.11.d ~~1.~~ Verifying that the pressure drop across the combined HEPA filters, pre-filters and charcoal adsorber banks is less than 8 ~~4~~ inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%. 2.26 L8.2

5.5.11.e ~~2.~~ Verifying that the heaters dissipate at least 0.5 kW when tested in accordance with ANSI N510-1980.

5.5.11a ~~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 N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

5.5.11.b ~~f.~~ After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

5.5.11 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequency. A.23

~~*ANSI N509-1980 is applicable for this specification.~~ LA.29

Chapter 5.0
(3.7.11/5.0/3.7.12)

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

A.1

LA.25

5.5.11.a ~~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 28,600 cfm ± 10%.

5.5.11.c ~~2.~~ Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

5.5.11.a ~~3.~~ Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.11.c ~~c.~~ After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

5.5.11 ~~d.~~ ~~At least once per 18 months by:~~ As specified in Regulatory Guide 1.52

5.5.11.d ~~1.~~ Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

ITS 5.0

ITS 3.7.11

2. Verifying that on a Control Room Essential Filtration Actuation Signal and on a SIAS, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8-inch Water Gauge relative to adjacent areas during system operation at a makeup flow rate to the control room of less than or equal to 1000 cfm.

ITS 3.7.11

ITS 3.7.12

4. Verifying that the emergency chilled water system will maintain the control room environment at a temperature less than or equal to 80°F for a period of 30 minutes.

ITS 3.7.12

ITS 5.0

*ANSI N509-1980 is applicable for this specification. LA.29

Chapter 5.0

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued) (A.1)

5.5.11.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 N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

5.5.11.b ~~f.~~ After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

5.5.11 — The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequency. — A.23

Chapter 5.0
(3.7.13/5.0)

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

A.1

A.25

5.5.11.a ~~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 6000 cfm ± 10%.

5.5.11.c ~~2.~~ Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

5.5.11.a ~~3.~~ Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.11.c ~~c.~~ After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

5.5.11 ~~d.~~ ~~At least once per 18 months by:~~ As specified in Regulatory Guide 1.52

5.5.11.d ~~1.~~ Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

ITS 5.0

ITS 3.7.13 2. Verifying that the system starts on an SIAS test signal.

ITS 5.0

5.5.11.a ~~e.~~ After each complete or partial replacement of an 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 N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

5.5.11.b ~~f.~~ After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

5.5.11 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequency. A.23

5.5.11 *ANSI N509-1980 is applicable for this specification.

5.5.11.e M.5 Verify that the heaters dissipate > 19 kW when tested in accordance with ASME N510-1980.

CHAPTER 5.0
(3.8.3/5.0)

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

DIESEL FUEL OIL STORAGE SYSTEM - DIESEL FUEL OIL REQUIREMENTS

LIMITING CONDITION FOR OPERATION

3.8.1.3.1 Each diesel fuel oil storage system shall be within its limits.

APPLICABILITY: When the associated EDG is required to be OPERABLE.

ACTION:

a. With either EDG fuel oil storage system < 80% indicated fuel level but ≥ 71% indicated fuel level, restore the fuel oil level to within its limit within 48 hours or declare the associated EDG inoperable.[1]

b. With either EDG fuel oil storage system with stored fuel oil viscosity not within limits, restore fuel oil to within limits within 30 days or declare the associated EDG inoperable.[1]

c. With either EDG fuel oil storage system with stored fuel oil water and sediment not within limits, immediately declare the associated EDG inoperable.[1]

[1] A separate condition entry is allowed for each EDG.

standards as referenced in the UFSAR. LA.28

SURVEILLANCE REQUIREMENTS

ITS 3.8.3 4.8.1.3.1.1 At least once per 31 days, verify that the fuel level in the fuel storage tank is within its limits.

ITS 5.0

5.5.13.c ~~4.8.1.3.1.2~~ At least once per 92 days, verify that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D4176-82 is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water and sediment.

SR 3.8.3.5

INSERT ITS 5.5.13.c A.24

INSERT 5.5.13, 5.5.13.a, 5.5.13.b M.4

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5.5.2 ~~3/4.11~~ RADIOACTIVE EFFLUENTS Explosive Gas and Storage Tank Radioactivity Monitoring Program

5.5.12.c ~~3/4.11.1 LIQUID HOLDUP TANKS~~

LIMITING CONDITION FOR OPERATION A.21

5.5.12.c ~~3.11.1.~~ The liquid quantity of radioactive material contained in each outside ~~temporary~~ tank and the reactor makeup water tank shall be limited to less than ~~or equal to 60 curies, excluding tritium and dissolved or entrained noble gases.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any outside temporary tank or the reactor makeup water 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.~~

LA.26

A.22

5.5.12 ~~b.~~ The provisions of ~~Specifications 3.0.3 and 3.0.4~~ SR 3.0.2 and SR 3.0.3 are ~~not~~ applicable.

SURVEILLANCE REQUIREMENTS A.21

5.5.12.c ~~4.11.1.~~ The liquid ~~quantity~~ of radioactive material contained in each outside ~~temporary~~ tank and the reactor makeup water tank shall be determined to be within the above limit ~~by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.~~ LA.26

A.20

the amount that would result in concentrations less than the limits of 10CFR20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

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RADIOACTIVE EFFLUENTS

5.5.12 ~~3/4.11.2~~ EXPLOSIVE GAS MIXTURE — Explosive Gas and Storage Tank Radioactivity Monitoring Program

LIMITING CONDITION FOR OPERATION

5.5.12.a ~~3.11.2.~~ The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume. (LA.27)

APPLICABILITY: Whenever waste gas holdup system is in service.

ACTION:

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

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 6 hours.

5.5.12 ~~c.~~ The provisions of ~~Specifications 3.0.3 and 3.0.4~~ SR 3.0.2 and SR 3.0.3 are ~~not~~ applicable. (A.22)

SURVEILLANCE REQUIREMENTS

4.11.2 The concentration of oxygen in the waste gas holdup system shall be determined to be within the above limts by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.8. (LA.27)

INSERT ITS 5.5.12.a (A.20)

A.1

~~RADIOACTIVE EFFLUENTS~~

5.5.12 ~~3/4.11.3~~ ~~GAS STORAGE TANKS~~ Explosive Gas and Storage Tank Radioactivity Monitoring Program

LIMITING CONDITION FOR OPERATION

5.5.12.b ~~3.11.3.~~ The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 170,000 curies noble gases (considered as Xe-133) LA.26

~~APPLICABILITY: At all times.~~

~~ACTION:~~

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

5.5.12 ~~b.~~ The provisions of Specifications 3.0.3 and 3.0.4 are not applicable. SR 3.0.2 and SR 3.0.3 A.22

SURVEILLANCE REQUIREMENTS

5.5.12.b ~~4.11.3~~ The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added to the tank and the quantity of radioactivity contained in the tank is less than or equal to one-half of the above limit; otherwise, determine the quantity of radioactive material contained in the tank at least once per 24 hours during addition. LA.26

A.20

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

CHAPTER 5.0

~~FOR INFORMATION ONLY~~

SECTION 6.0

ADMINISTRATIVE CONTROLS

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5.0 ADMINISTRATIVE CONTROLS

5.1 ~~6.1~~ RESPONSIBILITY

5.1.1 ~~6.1.1~~ The Department Leader, Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. LA.1

5.1.2 ~~6.1.2~~ The Shift Supervisor, or during his absence from the Control Room, a designated individual ~~per Table 6.2-1,~~ shall be responsible for the Control Room command function. ~~A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.~~ A.2

5.2 ~~6.2~~ ORGANIZATION

5.2.1 ~~6.2.1~~ OFFSITE AND ONSITE ORGANIZATIONS

An offsite and an onsite organization shall be established for unit operation and corporate management. The offsite and onsite 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 from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the FSAR ~~and updated in accordance with 10 CFR 50.71(e).~~ A.1 INSERT 1 A3

5.2.1.c ~~b.~~ There shall be an individual executive position (Executive Vice President Nuclear) in the offsite organization having corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support in the plant so that continued nuclear safety is assured.

5.2.1.b ~~c.~~ There shall be an individual management position (Vice President, Nuclear Production) in the onsite organization having responsibility for overall unit safe operation and having control over those onsite resources necessary for safe operation and maintenance of the plant.

5.2.1.d ~~d.~~ Although the individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate management onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

5.2.2 ~~6.2.2~~ UNIT STAFF

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

INSERT FOR 6.2.1.a

INSERT 1

, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation

INSERT PAGE 6-1

CHAPTER 5.0

5.0 ADMINISTRATIVE CONTROLS (A.1)

5.2.2 UNIT STAFF (Continued)

~~b.~~ At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room. (LA.30)

5.2.2.c c. A radiation protection technician* shall be onsite when fuel is in the reactor.

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. (LA.2)

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the 3 other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. (LA.3)

5.2.2.d ~~6.2.2.1~~ The unit staff working hours shall be as follows:

Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

(LA.3)

5.2.2.c *The radiation protection technician and Fire Team 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.

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5.2.2 UNIT STAFF (Continued)

5.2.2.d ~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant 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 modifications, on a temporary basis, the following guidelines shall be followed (this excludes the STA and PVNGS Fire Department working hours):~~

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

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

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

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

LA.3

Any deviation from the ~~above~~ working hour guidelines shall be authorized in advance by personnel who are at the Director level or above, or their designees who are at the Department Leader level or above, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime in their respective groups shall be reviewed monthly by these authorized individuals or their designees to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

LA.3 M.1

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

LA.3

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
~~SS~~	~~1~~	~~1~~
~~SRO~~	~~1~~	~~None~~
~~RO~~	~~2~~	~~1~~
AO	2	1
~~STA~~	~~1~~	~~None~~

LA.1

5.2.2.a

~~SS - Shift Supervisor with a Senior Reactor Operators License~~
~~SRO - Individual with a Senior Reactor Operators License~~
~~RO - Individual with a Reactor Operators License~~
~~AO - Nuclear Operator I or II~~
~~STA - Shift Technical Advisor~~

LA.1

5.2.2.b

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 resotre 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.

LA.1

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 with a valid Senior Operator 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 Senior Operator or Operator license shall be designated to assume the Control Room command function.

Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m).

A.26

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5.0 ADMINISTRATIVE CONTROLS

6.2.3 INDEPENDENT SAFETY ENGINEERING DEPARTMENT (ISE)

FUNCTION

6.2.3.1 The ISE Department shall function to selectively examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISE Department shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.

RESPONSIBILITIES

6.2.3.3 The ISE Department shall be responsible for maintaining surveillance of selected plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

(LA4)

AUTHORITY

6.2.3.4 The ISE Department shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Director, Site Operations, and the Chairman, Offsite Safety Review Committee (OSRC).

RECORDS

6.2.3.5 Records of activities performed by the ISE Department shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Assurance or designated alternate.

5.2.2.f ~~6.2.4~~ SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

5.3 ~~6.3~~ UNIT STAFF QUALIFICATIONS

5.3.1 ~~6.3.1~~ Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 and Regulatory Guide 1.8, September 1975, except for the Director, Site Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating

*Not responsible for sign-off function. (LA4)

A.1

5.0 ADMINISTRATIVE CONTROLS

5.3 ~~6.3~~ UNIT STAFF QUALIFICATIONS (Continued)

5.3.1 characteristics, including transients and accidents. An additional exception is that the Senior Reactor Operator (SRO) license requirement for the Operations Department Leader shall be met if either the Operations Department Leader or the Operations Supervisor holds a valid SRO license. The holder of the SRO license shall direct the licensed activities of the licensed operators. A4

5.2.2.e

INSERT 5.3.2

A.26

~~6.4 TRAINING~~

~~6.4.1 A training program for the unit staff shall be maintained under the direction of the Director, Nuclear Training and shall meet or exceed the requirements of Section 5.0 of ANSI/ANS 3.1, 1978 and 10 CFR 55. The program shall include familiarization with relevant industry operational experience.~~ LA.5

~~6.5 REVIEW AND AUDIT~~

~~6.5.1 PLANT REVIEW BOARD (PRB)~~

~~FUNCTION~~

~~6.5.1.1 The Plant Review Board shall function to advise the Vice President Nuclear Production or his designee on all matters related to nuclear safety.~~

~~COMPOSITION~~

~~6.5.1.2 The PRB shall be composed of at least seven members from the Palo Verde management staff. These positions will be designated by the Vice President Nuclear Production or his designee in Administrative Procedures.~~ LA.6

~~The Vice President Nuclear Production or his designee shall designate the Chairman and designated alternate in writing. The Chairman and designated alternate may be from outside the members provided that they meet ANSI/ANS 3.1, 1978.~~

~~ALTERNATES~~

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

~~MEETING FREQUENCY~~

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

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QUORUM

6.5.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or a designated alternate and a majority of the members including alternates.

RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

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

b. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Offsite Safety Review Committee (OSRC).

c. Review of REPORTABLE EVENTS.

d. Review of unit operations to detect potential nuclear safety hazards. LA.6

e. Performance of special reviews, investigations or analyses and reports thereon as requested by the Vice President-Nuclear Production or PRB Chairman.

f. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

AUTHORITY

6.5.1.7 The PRB shall:

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

b. Provide written notification within 24 hours to the Executive Vice President Nuclear, Vice President Nuclear Production and OSRC of disagreement between the PRB and the Vice President Nuclear Production; however, the Vice President Nuclear Production shall have responsibility for resolution of such disagreements.

RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Executive Vice President Nuclear, Vice President Nuclear Production, and OSRC.

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5.0 ADMINISTRATIVE CONTROLS

6.5.2 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.2.1 The Vice President Nuclear Production or his designee shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Director, System Engineering or his designee as previously designated by the Vice President Nuclear Production. Test results shall be approved by the Director, System Engineering or his designee. (LA.6)

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. (A.5) Proposed modifications to nuclear safety-related structures, systems and components shall be approved (his designee) prior to implementation by the Department Leader, Operations, or by the Director, Site Operations as previously designated by the Vice President Nuclear Production. (5.1.1)

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be identified in station procedures. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel. (LA.6)

(5.1.1) ~~6.5.2.5~~ Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Vice President Nuclear Production (Department Leader, Operations) or his designee. (L.1)

6.5.2.6 Not used.

6.5.2.7 Not used. (A.1)

6.5.2.8 The Director, Site Radiation Protection shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering the evaluation, recommendations and disposition of the corrective action to prevent recurrence. (LA.6)

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6.5.3 OFFSITE SAFETY REVIEW COMMITTEE (OSRC)

FUNCTION

6.5.3.1 The OSRC shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices .

LA.6

COMPOSITION

6.5.3.2 The OSRC shall be composed of the OSRC Chairman and a minimum of four OSRC members. The Chairman and members are designated by the Executive Vice President, Nuclear and shall have the qualifications that meet the requirements of Section 4.7 of ANSI/ANI 3.1; 1978.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the OSRC Chairman to provide expert advice to the OSRC.

REVIEW

6.5.3.4 The OSRC shall review:

a. The safety evaluations program and its implementation for (1) changes to procedures, equipment, systems or facilities within the power block, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

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REVIEW (Continued)

b. Proposed changes to procedures, equipment, systems or facilities within the power block which involve an unreviewed safety question as defined in 10 CFR 50.59;

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

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

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

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

g. All REPORTABLE EVENTS requiring 24 hours written notification;

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; and

LA.6

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the OSRC. These audits shall encompass:

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

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

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the OSRC or the Executive Vice President Nuclear.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

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AUDITS (Continued)

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

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

k. The performance of activities required by the Operations Quality Assurance Criteria Manual to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.3.6 The OSRC shall report to and advise the Executive Vice President Nuclear on those areas of responsibility specified in Specifications 6.5.3.4 and 6.5.3.5. LA.6

RECORDS

6.5.3.7 Records of OSRC activities shall be prepared and maintained. Report of reviews and audits shall be forwarded to the Executive Vice President Nuclear with distribution to the management positions responsible for the areas audited.

ALTERNATES

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

MEETING FREQUENCY

6.5.3.9 The OSRC shall meet at least once per six months.

QUORUM

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

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~~6.6 REPORTABLE EVENT ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50, 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 PRB, and the results of this review shall be submitted to the Chairman, Offsite Safety Review Committee and the Vice President-Nuclear Production.~~

LA.7

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5.0 ADMINISTRATIVE CONTROLS

6.7 SAFETY LIMIT VIOLATION

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

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President Nuclear Production, Director, Site Operations and Chairman of the OSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. 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.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the OSRC and the Vice President Nuclear Production within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

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5.4 ~~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, and those required for implementing the requirements of NUREG-0737.

5.4.1.b

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety-related equipment.~~ A.6

~~d. Not used.~~

~~e. Not used.~~ A.1

INSERT 5.4.1.e M.2

5.4.1.d ~~f.~~ Fire Protection Program implementation.

5.4.1.f ~~g.~~ Modification of Core Protection Calculator (CPC) Addressable Constants. INSERT 1 M.3

~~NOTES: (1) Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.~~ LA.8

5.4.1.f ~~(2)~~ Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limit values shall not be implemented without prior NRC approval.

INSERT FOR 6.8.1.g

INSERT 1

These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

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5.0 ADMINISTRATIVE CONTROLS

5.4.3 PROCEDURES AND PROGRAMS (Continued)

~~h. PROCESS CONTROL PROGRAM implementation.~~ (A.6)

~~i. OFFSITE DOSE CALCULATION MANUAL implementation~~ (A.7)

5.4.1.c ~~j.~~ 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.9)

~~k. Pre-planned Alternate Sampling Program implementation~~ (A.6)

~~l. Secondary water chemistry program implementation.~~

~~NOTE: The licensee shall perform a secondary water chemistry monitoring and control program that is in conformance with the program discussed in Section 10.3.4.1 of the CESSAR FSAR or another NRC approved program.~~ (A.7)

~~m. Post-Accident Sampling System implementation.*~~

~~n. Settlement Monitoring Program implementation.~~

~~NOTE: The licensee shall maintain a settlement monitoring program throughout the life of the plant in accordance with the program presented in Table 2.5-18 of the PVNGS FSAR or another NRC approved program.~~

~~o. CEA Reactivity Integrity Program implementation~~

~~NOTE: The licensee shall perform, after initial fuel load or after each reload, either a CEA symmetry test or worth measurements of all full-length CEA groups to address Section 4.2.2 of the PVNGS SER dated November 11, 1981.~~ (LA.10)

~~p. Fuel Assembly Surveillance Program Implementation~~

~~NOTE: The licensee shall perform a fuel assembly surveillance program in conformance with the program discussed in Section 4.2.4 of the PVNGS SER dated November 11, 1981.~~

~~6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5 and approved prior to implementation. Programs, administrative control procedures and implementing procedures shall be approved by the Vice President-Nuclear Production, or designated alternate who is at supervisory level or above. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.~~

~~6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:~~ (LA.11)

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

~~b. The change is approved by two members of the plant supervisory staff, at least one of whom is a Shift Supervisor or Control Room Supervisor with an SRO on the affected unit.~~

~~c. The change is documented, reviewed in accordance with Specification 6.5.2 and approved by the cognizant department head, as designated by the Vice President-Nuclear Production, within 14 days of implementation.~~

~~*Not required until prior to exceeding 5% of RATED THERMAL POWER.~~ (A.7)

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5.0 ADMINISTRATIVE CONTROLS

5.5 PROCEDURES AND PROGRAMS (Continued)

5.5 ~~6.8.4~~ The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the OSRC at least once per 24 months: LA.12

5.5.2 ~~a.~~ Primary Coolant Sources Outside Containment

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. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sample return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

5.5.2.a ~~(1)~~ Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b ~~(2)~~ Integrated leak test requirements for each system at refueling cycle intervals or less.

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.13

(1) Training of personnel,

(2) Procedures for monitoring, and

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

5.5.10 c. Secondary Water Chemistry

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

5.5.10.a ~~(1)~~ Identification of a sampling schedule for the critical variables and control points for these variables,

5.5.10.b ~~(2)~~ Identification of the procedures used to measure the values of the critical variables,

5.5.10.c ~~(3)~~ Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

5.5.10.d ~~(4)~~ Procedures for the recording and management of data,

A.1

5.0 ADMINISTRATIVE CONTROLS

5.5 PROCEDURES AND PROGRAMS (Continued)

5.5.10.e ~~(5)~~ Procedures defining corrective actions for all off-control point chemistry conditions, and

5.5.10.f ~~(6)~~ 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.

~~d. Backup Method for Determining Subcooling Margin~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~

~~(1) Training of personnel, and~~

~~(2) Procedures for monitoring.~~

LA.13

5.5.3 e. 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:

5.5.3.a ~~(1)~~ Training of personnel,

5.5.3.b ~~(2)~~ Procedures for sampling and analysis,

5.5.3.c ~~(3)~~ Provisions for maintenance of sampling and analysis equipment.

~~f. Spray Pond Monitoring~~

~~A program which will identify and describe the parameters and activities used to control and monitor the Essential Spray Pond and Piping. The program shall be conducted in accordance with station manual procedures.~~

LA.13

5.5.4 ~~g.~~ 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:

functional capability — A.1

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, 10 times the concentration values in

A.1

5.0 ADMINISTRATIVE CONTROLS

5.5 PROCEDURES AND PROGRAMS (Continued)

to 10 CFR 20.1001 - 20.2402 A.27

5.5.4.b ~~(2)~~ Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ~~10 CFR Part 20~~, Appendix B, Table ~~II~~ 2, Column 2 ~~,~~

5.5.4.c ~~(3)~~ Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.~~106~~ 1302 and with the methodology and parameters in the ODCM, A.8

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,~~ A.1

5.5.4.f ~~(6)~~ Limitations on the ~~operability~~ functional capability 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 from the site to areas at or beyond the SITE BOUNDARY ~~conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1~~, A.27

INSERT 1

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 Part 50,

5.5.4.j ~~(10)~~ 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 Part 190. A.27

INSERT FOR 6.8.4.g(7)

INSERT 1

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and (A.27)

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

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5.0 ADMINISTRATIVE CONTROLS

h. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides 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,

(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.

LA.14

5.6 ~~6.9~~ REPORTING REQUIREMENTS

ROUTINE REPORTS

in accordance with 10CFR 50.4

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 Regional Administrator of the Regional Office of the NRC unless otherwise noted.~~ A.9

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) 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. LA.15

INSERT ITS 5.5.13, 5.5.14, and 5.5.15 M.4

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5.0 ADMINISTRATIVE CONTROLS

5.6 REPORTING REQUIREMENTS (Continued)

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 required in license conditions based on other commitments shall be included in this report. LA.15

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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

5.6.1 ~~6.9.1.4~~ Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted by April 30 (A.10) ~~within the first calendar quarter~~ of each year. ~~The initial report shall be submitted within the first calendar quarter of the year following initial criticality.~~ A.1

5.6.1 ~~6.9.1.5~~ Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/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. 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% 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.

Annual reports shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included: (1) Reactor power history starting 48 hours prior the 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 LA.16

5.6.1, 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.

5.6.1 **This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

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5.0 ADMINISTRATIVE CONTROLS

5.6 ANNUAL REPORTS (Continued)

~~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.~~ LA.16

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 safety valves, shall be submitted on a monthly basis ~~to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC,~~ A.11 no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

5.6.2 ~~6.9.1.7~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. A.10

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

5.6.3 ~~6.9.1.8~~ The Annual Radioactive Effluent Release 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 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.

5.6.2 ~~*~~ A single submittal may be made for a multi-unit station.

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 release of radioactive material from each unit.

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5.0 ADMINISTRATIVE CONTROLS

5.6.5 CORE OPERATING LIMITS REPORT

5.6.5.a ~~6.9.1.9~~ 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:

a. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.1
b. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
c. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
d. Boron Dilution Alarms for Specification 3.1.2.7
e. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
f. Regulating CEA Insertion Limits for Specification 3.1.3.6
g. Part Length CEA Insertion Limits for Specification 3.1.3.7
h. Linear Heat Rate for Specification 3.2.1
i. Azimuthal Power Tilt - T_q for Specification 3.2.3
j. DNBR Margin for Specification 3.2.4
k. Axial Shape Index for Specification 3.2.7
l. Boron Concentration (Mode 6) for Specification 3.9.1

5.6.5.b ~~6.9.1.10~~ The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.1.2, Shutdown Margin Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

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CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

5.6.5.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 analysis limits) of the safety analysis are met.

5.6.5.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.11)

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~~SPECIAL REPORTS~~

~~6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.~~ A.12

~~6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with 10 CFR 50.73.~~ LA.17

~~6.10 RECORD RETENTION~~

~~In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.~~

~~6.10.1 The following records shall be retained for at least 5 years:~~

- a. ~~Records and logs of unit operation covering time interval at each power level.~~
- b. ~~Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.~~
- c. ~~All REPORTABLE EVENTS submitted to the Commission.~~
- d. ~~Records of surveillance activities, inspections and calibrations required by these Technical Specifications.~~
- e. ~~Records of changes made to the procedures of Specification 6.8.1.~~
- f. ~~Records of radioactive shipments.~~
- g. ~~Records of sealed source and fission detector leak tests and results.~~ LA.18
- h. ~~Records of annual physical inventory of all sealed source material of record.~~

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

- a. ~~Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.~~
- b. ~~Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.~~
- c. ~~Records of radiation exposure for all individuals entering radiation control areas.~~
- d. ~~Records of gaseous and liquid radioactive material released beyond the SITE BOUNDARY.~~
- e. ~~Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.~~
- f. ~~Records of reactor tests and experiments.~~
- g. ~~Records of training and qualification for current members of the unit staff.~~
- h. ~~Records of inservice inspections performed pursuant to these Technical Specifications.~~

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RECORD RETENTION (Continued)

i. Records of quality assurance activities required by the QA Manual not listed in Section 6.10.1.

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

k. Records of PRB meetings and of OSRC activities.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

LA.18

m. Records of audits performed under the requirements of Specifications 6.5.3.5 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

q. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

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

LA.19

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5.0 ADMINISTRATIVE CONTROLS

5.7 ~~6.12~~ HIGH RADIATION AREA 1601 A.8

5.7.1 ~~6.12.1~~ In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 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 Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. A radiation protection qualified individual (i.e., 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 facility Radiation Protection Section Leader or his designated alternate in the REP.

5.7.2 ~~6.12.2~~ In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems**, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.7.3

5.7.2

5.7.1 *Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

30 centimeters

5.7.2 **Measurement made at 18 inches from source of radioactivity. LB.1

CHAPTER 5.0

5.0 ADMINISTRATIVE CONTROLS

A.1

~~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.q. This documentation shall contain:~~

~~(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 PRB and the approval of the Director, Site Radiation Protection.~~

LA.14

5.5.1 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

5.5.1.a a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.q. This documentation shall contain: LA.18

(1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and A.8

(2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 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. 1302

5.5.1.b b. Shall become effective after review and acceptance by the PRB and the approval of the Director, Site Chemistry. LA.20

5.5.1.c c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as 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.

~~6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*~~

~~6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):~~

~~Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:~~

~~1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.~~

LA.14

~~*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.~~

CHAPTER 5.0

(A.1)

2) ~~Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;~~

3) ~~A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;~~

4) ~~An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;~~

5) ~~An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;~~

6) ~~A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and~~

7) ~~An estimate of the exposure to plant operating personnel as a result of the change.~~

LA.14

5.5.16 ~~6.16~~ CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR Part 50.54(o) and 10 CFR Part 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 and ANSI/ANS-56.8-1994.

The containment design pressure is 60psig

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~49.5~~ 52.0 psig.

A.25

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

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.6\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

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

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 14.5 ~~(± 0.5)~~ psig.

A.13

DISCUSSION OF CHANGES
CHAPTER 5.0

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CE) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the PVNGS Current Technical Specifications (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 6.1.2 requires that the Vice President-Nuclear Production issue a management directive to all station personnel on an annual basis that the Shift Supervisor is responsible for the Control Command function. ITS does not include the management directive. CTS and ITS state who is responsible for the control room command function. The UFSAR also delineates the responsibilities of the Shift Supervisor. CTS 6.1.2 serves only as a reminder to personnel as to who is in charge. No where else in TS is a management directive required to remind personnel of a TS requirement. Since the CTS responsibility requirement is not being changed, this deletion is administrative, with no impact on the margin of safety. This change is consistent with NUREG-1432.

A.3 CTS requires that the organizational charts in the UFSAR be updated in accordance with 10 CFR 50.71(e). ITS does not include this requirement. 10 CFR 50.71(e) includes the requirements for updating the UFSAR. Therefore, it is not necessary to repeat this requirement in TS. This is an administrative change with no impact on safety. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.4 CTS 6.3.1 states that the holder of the SRO license, Operations Department Leader or Operations Supervisor, shall direct the licensed activities of the licensed operators. This requirement is not included in ITS. ITS 5.1.1 states that the "Department Leader Operations shall be responsible for overall unit operation." ITS 5.2.1.a requires that "lines of authority, responsibility, and communication shall be established..." for "...all operating organization positions." and "these requirements shall be documented in the UFSAR." Therefore, this is an administrative change that does not affect nuclear safety. This change is consistent with NUREG-1432.

A.5 CTS 6.5.2.3 requires that proposed modifications to nuclear-safety related structures, systems, and components be approved prior to implementation by the Department Leader, Operations; or by the Director, Site Operations as previously designated by the Vice President Nuclear Production. ITS 5.1.1 requires that the Department Leader, Operations or his designee approve modifications to systems or equipment that affect nuclear safety prior to implementation. The Department Leader, Operations reports to the Director, Site Operations. Therefore, this additional option for approval authority does not need to be specifically discussed in the specification and is being removed. Since the requirements are not changed, this is an administrative change and does not impact the margin of safety. This change is consistent with NUREG-1432.

A.6 CTS 6.8.1 provides requirements for written procedures. The procedures required by CTS 6.8.1.b, 6.8.1.c, 6.8.1.h, and 6.8.1.k are not specifically detailed in ITS 5.4.1 since they are also are also required by CTS 6.8.1.a (ITS 5.4.1.a) which references Regulatory Guide 1.33. Therefore, it is not necessary to identify each type of procedure separately. Since the requirements are not changed, this is an administrative change and does not impact the margin of safety. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.7 CTS 6.8.1.i, 6.8.1.l and 6.8.1.m require written procedures for ODCM implementation, secondary water chemistry program implementation and Post-Accident Sampling System implementation, respectively. ITS 5.4.1.e requires written procedures for all Programs identified in specification 5.5. The ODCM, secondary water chemistry program and Post-Accident Sampling System are included in the programs listed in specification 5.5. Therefore, it is not necessary to identify each type of procedure. Since the requirements are not changed, this is an administrative change and does not impact the margin of safety. This change is consistent with NUREG-1432.

A.8 CTS references 20.106, 20.203, and table II of 10 CFR 20. ITS references paragraph 20.1302, 20.1601, and table 2 respectively, which reflects the latest version of 10 CFR 20. The requirements identified in each specification are the same and this change is an administrative change that changes the references to the revised 10 CFR 20. This change is consistent with NUREG-1432.

A.9 CTS 6.8.1 requires that "In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator...." ITS requires that "The following reports shall be submitted in accordance with 10 CFR 50.4." 10 CFR 50.4 provides the NRC distribution requirements for report submittal. This change is consistent with NUREG-1432.

A.10 CTS 6.9.1.4 requires that annual reports be submitted within the first calendar quarter of each year. CTS 6.9.1.7 requires that the Annual Radiological Environmental Operating Report be submitted before May 1 of each year. ITS 5.6.1 requires that the Occupational Radiation Exposure Report (annual report) be submitted by April 30 of each year. ITS 5.6.2 requires that the Annual Radiological Environmental Operating Report be submitted by May 15 of each year. Report submittal is not required to assure safe operation of the plant. Additionally there is no requirement for the NRC to approve these reports. Therefore, this change in the submittal dates for these reports does not impact safe plant operation. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.11 CTS 6.9.1.6 AND 6.9.1.10 require reports to be submitted to offices of the NRC. The ITS requires submittal of reports in accordance with 10 CFR 50.4. 10 CFR 50.4 provides the NRC distribution requirements for report submittal. The ITS submittal of reports requirements are sufficient without including unnecessary details. This change does not impact the technical requirements and therefore is considered an administrative change. This change is consistent with NUREG-1432.

A.12 CTS 6.9.2 requires submittal of special reports to "the NRC within the time period specified for each report." ITS does not include this requirement. Each special report contains requirements for submittal. This change deletes duplicate requirements in the Technical Specifications and therefore is an administrative change. This change is consistent with NUREG-1432.

A.13 CTS 6.16.b.2) requires that the air lock testing be performed with the door pressurized to ≥14.5 ± 0.5 psig. ITS requires air lock testing be performed with the door pressurized to ≥14.5 psig. The "±0.5" is a redundant requirement and is being removed from the acceptance criteria. This change is consistent with NUREG-1432.

A.14 CTS 4.0.5 establishes requirements for testing ASME Code Class 1, 2, & 3 components. ITS 5.5.8 establishes the same requirements and adds a statement that testing include "applicable supports." This is an administrative change because Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda already specify that testing include applicable supports. This change is consistent with NUREG-1432.

A.15 CTS 4.0.5.d states that the inservice inspection program and testing activities must be performed in addition to other CTS surveillance requirements. This statement is not retained in the ITS because all TS surveillance requirements must be performed as required. Deletion of this statement is an administrative change. This change is consistent with NUREG-1432.

A.16 A clear statement that SR 3.0.3 is applicable to ITS 5.5.8 is added, since the ITS Applicability SRs are not normally applied to frequencies identified in the Administrative Controls Chapter of the Technical Specifications. The addition of this statement is an administrative change because it maintains an allowance that is available in CTS 4.0.5. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.17 An ASME biennial or 2 year testing interval with a 731 day frequency is added to the testing frequency list in CTS 4.0.5.b. This change adds a frequency already identified in the ASME code. Therefore his change is an administrative change that is consistent with NUREG-1432.

A.18 CTS 3.4.8.3 action e requires that "In the event that the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS transient, a Special Report shall be submitted to the Commission...within 30 days. ITS requires that all "challenges to the SCS suction line relief valves or pressurizer safety valves, shall be submitted on a monthly basis" in the monthly operating report. Therefore his change is an administrative change that is consistent with NUREG-1432.

A.19 A clear statement that SR 3.0.2 and SR 3.0.3 are applicable to ITS 5.5.6 is added because the ITS Applicability SRs are not normally applied to frequencies identified in the Administrative Controls Chapter of the Technical Specifications. The addition of this statement is an administrative change because it maintains an allowance that is available in the CTS for this testing. This change is consistent with NUREG-1432.

A.20 The Technical Specification requirements for the Liquid Holdup Tanks, Explosive Gas Mixture, and Gas Storage Tanks in CTS 3.11.1, 3.11.2, and 3.11.3, respectively are moved to ITS 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program. ITS 5.5.12 will contain all of the requirements for explosive gas mixture and the quantity of radioactivity in liquid and gaseous storage tanks. This is an administrative change since the requirements are being retained in the monitoring program. This change is consistent with NUREG-1432.

A.21 CTS 3.11.1, Liquid Holdup Tanks, establishes limits for radioactive material stored in outside unprotected storage tanks. ITS 5.5.12 replaces the word "temporary" with a precise description of what constitutes a temporary tank, i.e., all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system. This is an administrative change since the requirements are being retained in the monitoring program. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.22 A clear statement that SR 3.0.2 and SR 3.0.3 are applicable to ITS 5.5.12 is added because the ITS Applicability SRs are not normally applied to frequencies identified in the Administrative Controls Chapter of the Technical Specifications. The addition of this statement is an administrative change because it maintains an allowance that is available in CTS 3.11.1, 3.11.2, and 3.11.3. This change is consistent with NUREG-1432

A.23 A clear statement that SR 3.0.2 and SR 3.0.3 are applicable to ITS 5.5.11 is added because the ITS Applicability SRs are not normally applied to frequencies identified in the Administrative Controls Chapter of the Technical Specifications. The addition of this statement is an administrative change because it maintains an allowance that is available in the CTS for ventilation filter testing. This change is consistent with NUREG-1432.

A.24 CTS 4.8.1.3.1.2 requires that a sample from the fuel storage tank be checked for viscosity and sediment. ITS 5.5.13.c requires that a sample from the fuel storage tank be checked for particulate concentration. These are equivalent and are both checking for degradation of the fuel oil. Therefore his change is an administrative change that is consistent with NUREG-1432.

A.25 CTS 3.6.1.3.b and 6.16 lists the calculate peak containment internal pressure, P_a as 49.5 psig. ITS 5.5.16 lists P_a as 52.0 psig. This change is characterized as an administrative change, since this number change is being addressed in a separate TS change request and submittal.

A.26 The requirement that "Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m)" is added to CTS Table 6.2-1. ITS 5.3.2 is added to CTS 6.3.1. These changes ensure that there is no misunderstanding when complying with 10 CFR 55.4 requirements. These changes incorporate the recommendations of NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997). These changes are clarifications of the existing requirements. Therefore, this is an administrative change that does not affect nuclear safety. This change is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.27 CTS 6.8.4.g is revised to incorporate changes to 10 CFR 20 and 10 CFR 50.36a. These changes are intended to eliminate possible problems with implementation of the revised 10 CFR 20 requirements. These changes incorporate the recommendations of NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997). 10 CFR 20 allows implementation of the rule without having to make technical specification changes. Therefore, this is an administrative change that does not affect nuclear safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 6.2.2.C allows deviation from the overtime guidelines, but does not clearly require advance authorization. ITS 5.2.2.d, governing work hours adds the requirement that: "Any deviations...be authorized in advance...." This is a more restrictive change which is consistent with NUREG-1432.

M.2 ITS 5.4.1.e is added to require procedures for TS required programs. This is a more restrictive change which is consistent with NUREG-1432.

M.3 CTS 6.8.1.g requires written procedures for modifications to the core protection calculator addressable constants. ITS 5.4.1.f also includes this requirement and in addition requires that "These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation." This is a more restrictive change which is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.4 Three new programs are included in the ITS. These programs are:

- 5.5.13 Diesel Fuel Oil Testing Program
- 5.5.14 Technical Specification (TS) Bases Control Program
- 5.5.15 Safety Functions Determination Program (SFDP)

The Diesel Fuel Oil Testing Program is provided to delineate the requirements for diesel fuel oil testing. The TS Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the TS Bases. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the TS. This is a more restrictive change which is consistent with NUREG-1432.

M.5 ITS 5.5.11.e adds the requirement for testing the ESF pump room exhaust air cleanup system in accordance with ASME N510-1980. CTS does not include this requirement. This is a more restrictive change which is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS Table 6.2.2-1 provides the minimum shift crew requirements. ITS does not include this requirement. The minimum shift crew requirements for licensed operators and senior reactor operators are contained in 10 CFR 50.54 (k), (l), and (m) and do not need to be repeated in the ITS. These details are also contained in the UFSAR. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.2 CTS 6.2.2.d requires that all Core Alterations to be supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling are removed from ITS. ITS does not include these requirements. These requirements are contained in 10 CFR 50.54 (m)(2)(iv). The requirements specified In 10 CFR 50.54 cannot be changed without prior approval from NRC. The qualification of personnel observing or directly controlling the Core Alteration is also specified in the UFSAR. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.3 CTS 6.2.2.e contains Site Fire Team requirements. This requirement is in CTS to ensure appropriate manning of the site Fire Team to be capable of responding to fire emergency. The fire protection specifications have been removed from CTS in accordance with Generic Letter 88-12. ITS does not include these requirements. Therefore, the personnel requirements are relocated to the UFSAR. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.4 CTS 6.2.3 provides requirements for Independent Safety Engineering Department (ISE). ITS does not include this requirement. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the Quality Assurance Program (QAP) description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

LA.5 CTS 6.4.1 identifies training requirements. ITS does not include these specific requirements. ITS 5.3, Unit Staff Qualifications, provides requirements to ensure adequate, competent staff in accordance with ANSI/ANS 3.1-1978 and Regulatory Guide 1.8, September 1975. ITS 5.2 details unit staffing requirements. Requirements for training and requalification of licensed positions are contained in 10 CFR 55. The training requirements are also discussed in the UFSAR, ensuring that training programs are properly maintained in accordance with PVNGS commitments and regulations. The requirements specified in either 10 CFR 55 or ITS cannot be changed without prior NRC approval. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.6 CTS 6.5 includes requirements for the Plant Review Board (PRB), technical review and control, and Offsite Safety Review Committee (OSRC). ITS does not include these requirements. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.7 CTS 6.6 provides specific actions for reportable events. ITS does not include these requirements. These requirements are included in 10 CFR 50.72 and 50.73. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.8 Note 1 to CTS 6.8.1.g requires prior PRB approval of modifications to CPC addressable constants based on information obtained through the plant computer - CPC data link. ITS does not include this requirement. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

LA.9 CTS 6.8.1.j requires that "the guidance of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975" be used. This requirement is not included in ITS. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.10 CTS 6.8.1.n, 6.8.1.o, and 6.8.1.p require programs that are described both in the CTS and UFSAR. These programs are not included in ITS. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.11 CTS 6.8.2 and 6.8.3 provide procedure review and approval requirements, and temporary changes to procedure requirements. These requirements are not included in ITS. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

LA.12 CTS 6.8.4 requires programs to "be audited under the cognizance of the OSRC at least once per 24 months." ITS does not include this requirement. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.13 CTS 6.8.4.b, 6.8.4.d, and 6.8.4.f require that programs be established for "In-Plant Radiation Monitoring", "Backup Method for Determining Subcooling Margin"; and "Spray Pond Monitoring." These requirements are not included in CTS. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.14 CTS 6.8.4.h, 6.13, and 6.15 provide requirements for the Radiological Environmental Monitoring Program, the Process Control Program, and Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems. These requirement are not included in ITS. These requirements are being relocated to the TRM. Changes to the TRM will be governed by the provisions 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the TRM is acceptable and is consistent with NUREG-1432

LA.15 The details associated with CTS 6.9.1.1, 6.9.1.2, and 6.9.1.3, Startup Report, are relocated to the UFSAR. ITS does not include this requirement. The Startup Report provides the NRC a mechanism to review the appropriateness of licensee activities after the fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.16 CTS 6.9.1.5 provides detailed requirements for the information included in the annual report. These requirements are not included in ITS. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore are being relocated to the TRM. Any changes to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements the TRM is acceptable and is consistent with NUREG-1432.

LA.17 CTS 6.9.3 requires reporting of fire protection program violations in accordance with 10 CFR 50.73. This requirement is not included in ITS. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.18 CTS 6.10 provides requirements for record retention. ITS does not include this requirement. This requirement is not included in ITS. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.19 CTS 6.11 provides details for the "Radiation Protection Program." The requirement to have procedures to implement 10 CFR 20 is contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed in 10 CFR 20.1101(c). This requirement is not included in ITS. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.20 CTS 6.14.b requires PRB review and acceptance of changes to the ODCM prior to the changes becoming effective. ITS does not include this requirement. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the QAP description in the UFSAR. Any changes to the requirements in the QAP description in the UFSAR will be governed by the provisions of 10 CFR 50.54(a). This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the QAP description in the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.21 CTS 4.0.5.a references specific 10 CFR 50 and ASME Code requirements governing performance of the inservice inspection and testing. These requirements are not included in ITS. The references duplicate 10 CFR 50.55a, which requires the implementation of ASME, Section XI and applicable addenda, for inservice inspection and testing of ASME Code Class 1, 2, and 3 components, pumps and valves. These specific requirements are maintained in the Inservice Testing Program. CTS 4.0.5 also refers to inservice inspection of ASME Code Class 1, 2, and 3 components shall be applicable per CTS 4.0.5.a. Any changes to the requirements in the Inservice Testing Program will be governed by the provisions of 10 CFR 50.55a and 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the Inservice Testing Program is acceptable and is consistent with NUREG-1432.

LA.22 CTS 4.5.2.e.4 requires verification that the total measured leakage from ECCS piping and components is less than 1 gpm when pressurized to at least 40 psig. ITS 5.5.2 requires that the program to minimize leakage from those portions of systems outside containment include integrated leak test requirements for each system. The integrated leak test requirements are being relocated to the TRM. Any changes to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the TRM is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.2 CTS 3.6.1.6 requires that the structural integrity of the containment vessel be maintained in Modes 1 through 4 and provides specific actions if the structural integrity is below the acceptance criteria. ITS does not include these requirements. ITS 5.5.6 establishes a program that meets the requirements of Regulatory Guide 1.35. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the TRM. Any changes to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the TRM is acceptable and is consistent with NUREG-1432.

LA.24 CTS 4.6.1.6.1, 4.6.1.6.2, 4.6.1.6.3, 4.6.1.6.4, and 4.6.1.6.5 provide detailed surveillance and reporting requirements for the structural integrity of the containment vessel. ITS does not include these detailed requirements. These specific requirements are maintained in the Pre-Stressed Concrete Containment Tendon Surveillance Program. Any changes to the requirements in the Pre-Stressed Concrete Containment Tendon Surveillance Program will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the Pre-Stressed Concrete Containment Tendon Surveillance Program is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.25 The Ventilation Filter Testing Program for the Control Room Essential Filtration System, ESF Pump Room Air Exhaust Cleanup System, and Fuel Building Essential Ventilation System in CTS 4.6.4.3, 4.7.7, and 4.7.8, respectively, are moved to the Ventilation Filter Testing Program. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the Ventilation Filter Testing Program. Any changes to the requirements in the Ventilation Filter Testing Program will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the Ventilation Filter Testing Program is acceptable and is consistent with NUREG-1432.

LA.26 Details of the method for implementing the Liquid Holdup Tanks and Gas Storage Tanks in CTS 3.11.1 and 3.11.3, are moved to the TRM. Changes to the TRM are controlled in accordance with the 10 CFR 50.59. These requirements are not required to determine the OPERABILITY of a system, component, or structure and therefore are being relocated to the TRM. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. These requirements are not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of these requirements to the TRM is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.27 CTS 3.11.2 establishes a maximum limit of 4% by volume for the oxygen concentration in the main condenser offgas holdup system, including Applicability and Action requirements. CTS 4.11.2 establishes a requirement for the continuous monitoring of the oxygen concentration in the waste gas holdup system to ensure that the appropriate limit is maintained. ITS 5.5.12.a requires that a limit for hydrogen and oxygen concentration be established that is appropriate based on design and that a surveillance program be used to ensure that the appropriate limits are maintained. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the TRM. Any changes to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the TRM is acceptable and is consistent with NUREG-1432.

LA.28 CTS 4.8.1.3.1.2 requires that a sample of diesel fuel from the fuel storage tank be obtained in accordance with ASTM-D4176-82. ITS 5.5.13 requires that sampling be performed in accordance with applicable ASTM Standards as referenced in the UFSAR. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.29 CTS 3.6.4.3, 3.7.7, and 3.7.8 reference ANSI N509-1980 in the Surveillance Requirements. ITS 5.5.11 does not include this reference. This reference is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This reference is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this reference to the UFSAR is acceptable and is consistent with NUREG-1432.

LA.30 CTS 6.2.2.b requires that "at least one licensed reactor operator shall be in the Control Room when fuel is in the reactor" and that "when the reactor is in Modes 1, 2, 3, and 4 at least one licensed Senior Reactor Operator shall be in the Control Room." NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997) deletes this requirement from the NUREG because the requirements are redundant to 10 CFR 50.54(m)(2)(iii). These details are being relocated to the UFSAR. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.31 CTS 6.2.2.1.b provides specific working hour limits for plant staff. These details are also contained in the UFSAR. NRC proposed change TSB-011 (letter from C. I. Grimes, NRC to J. Davis, NEI, dated April 9, 1997) changes this requirement. The change replaces the specific working hour limits with the requirement to establish guidelines on working hours that ensure adequate shift coverage without routine heavy use of overtime. The wording in CTS 6.2.2.1.c is also changed to be consistent with this change. This requirement is not required to determine the OPERABILITY of a system, component, or structure and therefore is being relocated to the UFSAR. Any changes to the requirements in the UFSAR will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of regulatory control and is an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement is not required to be in the ITS to provide adequate protection of public health and safety. Therefore, relocation of this requirement to the UFSAR is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 6.5.2.5 requires that the Vice President Nuclear Production or his designee review proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications. ITS 5.1.1 requires that the Department Leader, Operations approve, prior to implementation, each proposed test or experiment that affect nuclear safety. This change is consistent with the requirement in CTS 6.5.2.3 that requires that the Department Leader, Operations shall approve, prior to implementation, proposed modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review is changed, it results in consistent requirements for review and approval. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - CTS CHANGES

LB.1 CTS 6.12.2 Note ** states that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS 5.7.2 (NUREG exception) to ensure that it is clear where the dose is measured. ITS 5.7.2 provides requirements "In addition" to the requirements in ITS 5.7.1. ITS 5.7.1 references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20.

TECHNICAL CHANGES - CTS CHANGES (continued)

LB.2 CTS 4.6.4.3.d.1 lists maximum pressure drop for the hydrogen purge cleanup system (HPCS) as 8.4 inches water gauge. ITS 5.5.11.d lists the maximum delta P (pressure drop) for the HPCS as 2.26 inches water gauge. As part of the design basis verification project, APS engineering determined that the maximum pressure drop for the HPCS is correctly identified as 2.26 inches water gauge. During the original licensing process the system flowrate was revised and lowered from 1000 scfm to 50 scfm. The maximum allowable pressure drop was not revised at that time to correlate to the lower system flowrate. The change in the maximum allowable pressure drop to the correct value of 2.26 inches of water gauge ensures that the design basis requirements are maintained for the HPCS. Administrative controls have been put into place to ensure that the testing requirements use the correct value for the maximum allowable pressure drop.

NO SIGNIFICANT HAZARDS CONSIDERATION
CHAPTER 5.0

ADMINISTRATIVE CHANGES
(ITS 5.0 Discussion of Changes Labeled A.1, A.2, A.3, A.4, A.5, A.6, A.7, A.8, A.9, A.10, A.11, A.12, A.13, A.14, A.15, A.16, A.17, A.18, A.19, A.20, A.21, A.22, A.23, A.24, A.25, A26, and A.27)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

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

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 5.0 Discussion of Changes Labeled A.1, A.2, A.3, A.4, A.5, A.6, A.7, A.8, A.9, A.10, A.11, A.12, A.13, A.14, A.15, A.16, A.17, A.18, A.19, A.20, A.21, A.22, A.23, A.24, A.25, A26, and A.27)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 5.0 Discussion of Changes Labeled M.1, M.2, M.3, M.4, and M.5)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. This particular NSHC is for the changes labeled "Technical Changes - More Restrictive" described in the specific Discussion of Changes listed above. The proposed changes incorporate more restrictive changes into the CTS by either making current requirements more stringent or adding new requirements which currently do not exist.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

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

> The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes listed above. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 5.0 Discussion of Changes Labeled M.1, M.2, M.3, M.4, and M.5) (continued)

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG-1432. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG-1432. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 5.0 Discussion of Changes Labeled LA.1, LA.2, LA.3, LA.4, LA.5, LA.6, LA.7, LA.8, LA.9, LA.10, LA.11, LA.12, LA.13, LA.14, LA.15, LA.16, LA.17, LA.18, LA.19, LA.20, LA.21, LA.22, LA.23, LA.24, LA.25, LA.26, LA.27, LA.28, LA.29, LA.30 and LA.31)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

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

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 5.0 Discussion of Changes Labeled LA.1, LA.2, LA.3, LA.4, LA.5, LA.6, LA.7, LA.8, LA.9, LA.10, LA.11, LA.12, LA.13, LA.14, LA.15, LA.16, LA.17, LA.18, LA.19, LA.20, LA.21, LA.22, LA.23, LA.24, LA.25, LA.26, LA.27, LA.28, LA.29, LA.30 and LA.31) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 5.0 Discussion of Changes Labeled L.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

L.1 CTS 6.5.2.5 requires that the Vice President Nuclear Production or his designee review proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications. ITS 5.1.1 requires that the Department Leader, Operations approve, prior to implementation, each proposed test or experiment that affect nuclear safety. This change is consistent with the requirement in CTS 6.5.2.3 that requires that the Department Leader, Operations shall approve, prior to implementation, proposed modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review is changed, it results in consistent requirements for review and approval. This change is consistent with NUREG-1432.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 5.0 Discussion of Changes Labeled L.1) (continued)

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

The proposed change involves a change from the CTS requirement that requires that the Vice President Nuclear Production or his designee review proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications. ITS Requires that the Department Leader, Operations approve, prior to implementation, each proposed test or experiment that affect nuclear safety. This change is consistent with the requirement in CTS that requires that the Department Leader, Operations shall approve, prior to implementation, proposed modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review is changed, it results in consistent requirements for review and approval, since ITS 5.5.1 states that the Department Leader, Operations is responsible for overall unit operation. This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it effect plant operation. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves a change from the CTS requirement that requires that the Vice President Nuclear Production or his designee review proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications. ITS Requires that the Department Leader, Operations approve, prior to implementation, each proposed test or experiment that affect nuclear safety. This change is consistent with the requirement in CTS that requires that the Department Leader, Operations shall approve, prior to implementation, proposed modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review is changed, it results in consistent requirements for review and approval, since ITS 5.5.1 states that the Department Leader, Operations is responsible for overall unit operation. This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. This change will not alter assumptions

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 5.0 Discussion of Changes Labeled L.1) (continued)

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

(continued)
made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed change involves a change from the CTS requirement that requires that the Vice President Nuclear Production or his designee review proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications. ITS Requires that the Department Leader, Operations approve, prior to implementation, each proposed test or experiment that affect nuclear safety. This change is consistent with the requirement in CTS that requires that the Department Leader, Operations shall approve, prior to implementation, proposed modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review is changed, it results in consistent requirements for review and approval, since ITS 5.5.1 states that the Department Leader, Operations is responsible for overall unit operation. This change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

TECHNICAL CHANGES - CTS CHANGES
(ITS 5.0 Discussion of Changes Labeled LB.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

LB.1 CTS 6.12.2 Note ** states that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS 5.7.2 (NUREG exception) to ensure that it is clear where the dose is measured. ITS 5.7.2 provides requirements "In addition" to the requirements in ITS 5.7.1. ITS 5.7.1 references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

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

The proposed change involves a change from the CTS requirement that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS to ensure that it is clear where the dose is measured. ITS provides requirements "In addition" to the requirements in 10 CFR 20 and references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20. This change does not result in any hardware changes or changes to plant operating practices nor does it effect plant operation. Therefore, this

TECHNICAL CHANGES - CTS CHANGES
(ITS 5.0 Discussion of Changes Labeled LB.1) (continued)

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

(continued)

change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves a change from the CTS requirement that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS to ensure that it is clear where the dose is measured. ITS provides requirements "In addition" to the requirements in 10 CFR 20 and references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20. The proposed change, ensures that equipment performance will not be changed. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed change involves a change from the CTS requirement that the dose measurement is 18 inches from the source of radioactivity. The revision to 10 CFR 20 changed this measurement from 18 inches to 30 centimeters. Therefore, since ITS references the new version of 10 CFR 20, this distance is being updated. This note has been added to ITS to ensure that it is clear where the dose is measured. ITS provides requirements "In addition" to the requirements in 10 CFR 20 and references the requirements of 10 CFR 20. This change ensures that the ITS requirements are consistent with 10 CFR 20. This change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. Therefore, this change does not result in a reduction in a margin of safety.

TECHNICAL CHANGES - CTS CHANGES
(ITS 5.0 Discussion of Changes Labeled LB.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

LB.2 CTS 4.6.4.3.d.1 lists maximum pressure drop for the hydrogen purge cleanup system (HPCS) as 8.4 inches water gauge. ITS 5.5.11.d lists the maximum delta P (pressure drop) for the HPCS as 2.26 inches water gauge. As part of the design basis verification project, APS engineering determined that the maximum pressure drop for the HPCS (based on vendor supplied information) should be reduced to 2.26 inches water gauge. The original pressure drop was based on a system higher flow rate. This change ensures that the design basis requirements are maintained by ITS.

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

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

The proposed change involves a change from the CTS requires that the maximum pressure drop for the hydrogen purge cleanup system (HPCS) be less than 8.4 inches water gauge. ITS lists the maximum delta P (pressure drop) for the HPCS as 2.26 inches water gauge. As part of the design basis verification project, APS engineering determined that the maximum pressure drop for the HPCS should be reduced to 2.26 inches water gauge. The HPCS pressure drop is based on the pressure drop through the mist eliminator, the heating coil, the two HEPA filters, and the carbon absorber. During the original licensing process, the system design flow rate was lowered from 1000 cfm to 50 cfm. However, the maximum allowable pressure drop was not dated following the change in system design flow rate. This change ensures that the design basis requirements are maintained by ITS. This change does not result in any hardware changes or changes to plant operating practices nor does it effect plant operation. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - CTS CHANGES
(ITS 5.0 Discussion of Changes Labeled LB.2) (continued)

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

The proposed change involves a change from the CTS requires that the maximum pressure drop for the hydrogen purge cleanup system (HPCS) be less than 8.4 inches water gauge. ITS lists the maximum delta P (pressure drop) for the HPCS as 2.26 inches water gauge. As part of the design basis verification project, APS engineering determined that the maximum pressure drop for the HPCS (based on vendor supplied information) should be reduced to 2.26 inches water gauge. The original pressure drop was based on a system higher flow rate. This change ensures that the design basis requirements are maintained by ITS. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3.-- Does the proposed change involve a significant reduction in a margin of safety?

The proposed change involves a change from the CTS requires that the maximum pressure drop for the hydrogen purge cleanup system (HPCS) be less than 8.4 inches water gauge. ITS lists the maximum delta P (pressure drop) for the HPCS as 2.26 inches water gauge. As part of the design basis verification project, APS engineering determined that the maximum pressure drop for the HPCS (based on vendor supplied information) should be reduced to 2.26 inches water gauge. The original pressure drop was based on a system higher flow rate. This change ensures that the design basis requirements are maintained by ITS. This change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. Therefore, this change does not result in a reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

These proposed TS changes have 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 changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed TS changes meet the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed changes involve changes to requirements with respect to inspection or Surveillance Requirements with:

(i) the proposed changes involve No Significant Hazards Consideration (refer to the No Significant Hazards Consideration Section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite since the proposed changes do not affect generation of any radioactive effluent not 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 changes meet 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 impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.