B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.223 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Records

ENO shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction - DELETED

E. Recirculation Loop Inoperable - DELETED

F. Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:

ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004.

REFUELING INTERVAL	REFUELING INTERVAL applies only to Inservice Code Testing Program surveillance tests. For the purpose of designating frequency of these code tests, a REFUELING INTERVAL shall mean at least once every 24 months.
REFUELING OUTAGE	REFUELING OUTAGE is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled outage; however, where such outages occur within 11 months of completion of the previous REFUELING OUTAGE, the required surveillance testing need not be performed until the next regularly scheduled outage.
SAFETY LIMIT	The SAFETY LIMITS are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.
SECONDARY CONTAINMENT INTEGRITY	SECONDARY CONTAINMENT INTEGRITY means that the reactor building is intact and the following conditions are met: 1. At least one door in each access opening is closed. 2. The standby gas treatment system is operable. 3. All automatic ventilation system isolation valves are operable or secured in the isolated position.
SIMULATED AUTOMATIC ACTUATION	SIMULATED AUTOMATIC ACTUATION means applying a simulated signal to the sensor to actuate the circuit in question.
SOURCE CHECK	A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.
SURVEILLANCE FREQUENCY	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. The SURVEILLANCE FREQUENCY establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 for single recirculation loop operation.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met within two hours the following actions shall be met:

2.2.1 Restore compliance with all Safety Limits, and

2.2.2. Insert all insertable control rods.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

4. During Run, Startup, and Hot Shutdown Modes with the LPCI system inoperable, restore the LPCI system to Operable status within 7 days and maintain both core spray systems and the diesel generators Operable. Otherwise, be in at least Cold Shutdown within 24 hours.

5. Two low pressure injection/spray subsystems shall be Operable during Cold Shutdown and Refuel Modes unless the reactor head is removed, the spent fuel pool gates are removed, and water level is at greater than or equal to elevation 114 foot, except as specified in 3.5.A.6.

6. During Cold Shutdown and Refuel Modes unless the reactor head is removed, the spent fuel pool gates are removed, and water level is at greater than or equal to elevation 114 foot:

   a. With one of the required low pressure injection/spray subsystems inoperable, restore the inoperable required low pressure injection/spray subsystem to Operable status within 4 hours. Otherwise, take immediate action to suspend activities with potential for draining the reactor vessel.

   b. With both of the required low pressure injection/spray subsystems inoperable, take immediate action to suspend activities with potential for draining the reactor vessel and restore 1 low pressure injection/spray subsystem to Operable status within 4 hours. Otherwise, take immediate action to restore secondary containment and one standby gas treatment system to Operable status and to restore isolation capability in each required secondary containment penetration flow path not isolated.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

1. c. Motor Operated Valve Operability — As Specified in 3.13

   d. Core Spray Header Δp Instrumentation

Check	Once/day
Calibrate	Once/3 months
Test Step	Once/3 months

2. This section intentionally left blank

3. LPCI system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle
b. Pump Operability.	When tested as specified in 3.13, verify that each LPCI pump delivers 4800 GPM at a head across the pump of at least 380 ft.
c. Motor Operated Valve Operability	As Specified in 3.13

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

5. All containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position.

Primary Containment Isolation Valves

2. b. In the event any automatic Primary Containment Isolation Valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

4. Combined main steam lines: 46 scfh @ 23 psig.

where,

P_a = 45 psig
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

4. NEI 94-01-1995. Section 9.2.3: The first Type A test performed after the May 25, 1995 Type A test shall be performed no later than May 25, 2010.

Primary Containment Isolation Valves

2. b. 1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. Test primary containment isolation valves:

1. Verify power operated primary containment isolation valve operability as specified in 3.13.

2. Verify main steam isolation valve operability as specified in 3.13.

LIMITING CONDITIONS FOR OPERATION

3.8 PLANT SYSTEMS (CONT)

2. Mechanical Vacuum Pump Isolation Instrumentation

LCO 3.8.2

Four channels of the Main Steam Line Radiation Monitoring System Radiation - High function for the mechanical vacuum pump shall be OPERABLE

APPLICABILITY:

Whenever any main steam isolation valve is open with steam flowing.

ACTIONS:

----------NOTE----------
Separate Condition Entry is allowed for each channel.

A. One or more required channels inoperable.

1. Restore channel to OPERABLE status within 24 hours.

OR

2. ----------NOTE----------
Not Applicable if inoperable channel is the result of an inoperable isolation valve.

Place channel or associated trip system in trip within 24 hours.

B. Required Action and associated Completion Time of Condition A not met.

OR

Mechanical vacuum pump isolation capability not maintained.

1. Isolate mechanical vacuum within 12 hours.

OR

2. Isolate Main Steam Lines within 12 hours.

OR

3. Be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.8 PLANT SYSTEMS (CONT)

2. Mechanical Vacuum Pump Isolation Instrumentation

----------NOTE----------
When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains mechanical vacuum pump isolation capacity.
---

1. Perform a CHANNEL CHECK every 12 hours.

2. Calibrate the trip units every 92 days.

3. Perform a CHANNEL CALIBRATION every 24 months. The allowable trip value shall be ≤ 5.5 x normal background.

4. Perform a LOGIC SYSTEM FUNCTIONAL TEST including isolation valve actuation every 24 months.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment (Cont)

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment Surveillance (Cont)

1. Verifying de-energization of the emergency buses and load shedding from the emergency buses.

2. Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequence, and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

   During performance of this surveillance verify that HPCI and RCIC inverters do not trip.

   The results shall be logged.

c. Once per operating cycle with the diesel loaded per 4.9.A.1.b verify that on diesel generator trip, secondary (offsite) AC power is automatically connected within 11.8 to 13.2 seconds to the emergency service buses and emergency loads are energized through the load sequencer in the same manner as described in 4.9.A.1.b.2.

   The results shall be logged.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4, and 3.9.B.5.

1. From and after the date that incoming power is not available from the startup or shutdown transformer, continued reactor operation is permissible under this condition for:

   a. 3 days with the startup transformer inoperable

   or

   b. 7 days with the shutdown transformer inoperable

   During this period, both diesel generators and associated emergency buses must remain operable.

2. From and after the date that incoming power is not available from both startup and shutdown transformers, continued operation is permissible, provided both diesel generators and associated emergency buses remain operable, all core and containment cooling systems are operable, and reactor power level is reduced to 25% of design.

3. From and after the date that one of the diesel generators or associated emergency bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specifications 3.4.B.1, 3.5.F.1, 3.7.B.1.c, 3.7.B.1.e, 3.7.B.2.c, and 3.7.B.2.e if Specification 3.9.A.1 and 3.9.A.2.a are satisfied.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

Auxiliary Electrical Equipment Surveillance (Cont)

3. Emergency 4160V Buses A5-A6 Degraded Voltage Annunciation System.

   a. Once each operating cycle, calibrate the alarm sensor.

   b. Once each 31 days perform a channel functional test on the alarm system.

   c. In the event the alarm system is determined inoperable under 3.b above, commence logging safety related bus voltage every 30 minutes until such time as the alarm is restored to operable status.

4. RPS Electrical Protection Assemblies

   a. Each pair of redundant RPS EPAs shall be determined to be operable at least once per 6 months by performance of an instrument functional test.

   b. Once per 18 months each pair of redundant RPS EPAs shall be determined to be operable by performance of an instrument calibration and by verifying tripping of the circuit breakers upon the simulated conditions for automatic actuation of the protective relays within the following limits:

Overvoltage	$\leq$ 132 volts
Undervoltage	$\geq$ 108 volts
Underfrequency	$\geq$ 57Hz

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ratio (MCPR) (Cont'd)

2. The operating limit MCPR values as a function of the τ are given in Table 3.3-1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ratio (MCPR) (Cont'd)

b. The average scram time to dropout of Notch 34 is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \quad \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n = number of surveillance tests performed to date in the cycle.

N_i = number of active control rods measured in the i^{th} surveillance test.

τ_i = average scram time to dropout of Notch 34 of all rods measured in the i^{th} surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time to dropout of Notch 34, 0.937 sec.

N_1 = total number of active control rods at BOC during the first surveillance test.

σ = standard deviation of the distribution for average scram insertion time to the dropout of Notch 34, 0.021 seconds.

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

Applicability:

Applies to ASME Code Class 1, 2 and 3 pumps and valves.

Objective:

To assure the operational readiness of ASME Code Class 1, 2, and 3 pumps and valves.

Specification:

A. Inservice Code Testing of Pumps and Valves

1. Based on the Facility Commercial Operation Date, Inservice Code Testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with the Inservice Code Testing Program.

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

Applicability:

Applies to the periodic testing requirements of ASME Code Class 1, 2 and 3 pumps and valves.

Objective:

To assess the operational readiness of ASME Code Class 1, 2, and 3 pumps and valves by performance of inservice tests.

Specification:

A. Inservice Code Testing of Pump and Valves

1. The ASME OM Code terminology for Inservice Test activities is as follows.

Code Terminology	Frequencies
Weekly	7 Days
Monthly	31 Days
Quarterly or 3 Mths	92 Days
Semiannually/ 6 Mths	184 Days
9 Months	276 Days
Yearly/Annually	366 Days
Biannual/2 Yrs	732 Days

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

3. The provisions in Definitions (1.0) for REFUELING INTERVAL, SURVEILLANCE FREQUENCY, and SURVEILLANCE INTERVAL are applicable to Code testing and to the above frequencies for performing Code testing activities.

4. Performance of Code testing shall be in addition to other specified Surveillance Requirements.

5. Nothing in the Inservice Code Testing Program shall supersede the requirements of Technical Specifications.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

5.1.2 The 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 an operational mode other than Cold Shutdown or Refueling, 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 Cold Shutdown or Refueling, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Pilgrim Station Final Safety Analysis Report (FSAR);

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer for Pilgrim 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 on site when fuel is in the reactor and an additional non-licensed operator shall be assigned when the reactor is in an operational mode other than Cold Shutdown or Refueling.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

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 an operational mode other than Cold Shutdown or Refueling, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. At least two licensed ROs shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips .

d. 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.i 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.

e. Higher grade licensed operators may take the place of lower grade licensed or unlicensed personnel.

f. An individual qualified in radiation protection procedures 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.

g. 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).

h. The operations manager or assistant operations manager shall hold a Senior Reactor Operator License.

i. An individual shall provide advisory technical support to the unit operations shift crew in the areas of engineering and accident assessment. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. This individual with a Senior Reactor Operator license may simultaneously serve a required SRO position.

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), and

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

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

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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 to unrestricted areas, conforming to 10 CFR 50, Appendix I;

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

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

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site boundary to areas at or beyond the site boundary conforming to the following:

1. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin, and

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

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released 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 FSAR Section C.3.4.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Technical Specifications (TS) Bases Control Program

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

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

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

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

2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 5.5.6b 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).

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection personnel) 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 $\leq$ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

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

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

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

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

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels $\geq$ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of an SRO on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by

(Continued)