THIS PAGE INTENTIONALLY BLANK

1.28 (Deleted)

1.29 (Deleted)

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

1.32 Shutdown Margin (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free,

b. The moderator temperature is 68° F, and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point (5 feet below minimum normal water level). The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operating during the major maintenance with the water level below the low-low level set point.

In addition to the requirement that at least one licensed Operator be in the control room when fuel is in the reactor, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux for IRM range 9 or lower.

   The IRM scram trip setting shall not exceed 38.4% of rated neutron flux for IRM range 10.

c. The reactor high pressure scram trip setting shall be ≤ 1080 psig.

d. The reactor water low level scram trip setting shall be no lower than –12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than –5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be ≥ 850 psig when the reactor mode switch is in the run position or the IRMs are on range 10.

g. The main-steam-line isolation valve closure scram setting shall be ≤ 10 percent of valve closure (stem position) from full open.

LIMITING CONDITION FOR OPERATION

3.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines connected to the reactor coolant system.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system.

Specification:

a. During power operating conditions whenever the reactor head is on, all reactor coolant system isolation valves on lines connected to the reactor coolant system shall be operable except as specified in "b" below.

b. In the event any isolation valve becomes inoperable the system shall be considered operable provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition, except as noted in Specification 3.1.1.e.

SURVEILLANCE REQUIREMENT

4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the periodic testing requirement for the reactor coolant system isolation valves.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system.

Specification:

The reactor coolant system isolation valves surveillance shall be performed as indicated below.

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

b. Additional surveillances shall be performed as required by Specification 6.5.4.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10 CFR 100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

To assure that periodic surveillances of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Specification:

Whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required, the primary containment leakage rate shall be limited to:

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. The primary containment leakage rates shall be demonstrated at test schedules and in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan as described in Specification 6.5.7.

b. The provisions of Specification 4.0.1 are not applicable, and the surveillance interval extensions are in accordance with the 10 CFR 50 Appendix J Testing Program Plan.

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 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 a designee shall approve, prior to implementation, each proposed test and experiment not addressed in the UFSAR or Technical Specifications, and each modification to systems or equipment that affect nuclear safety.

6.1.2 The Station Shift Supervisor – Nuclear (SSS) shall be responsible for the control room command function. During any absence of the SSS from the control room while the unit is in the power operating or hot shutdown conditions, 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 SSS from the control room while the unit is in the cold shutdown or refueling conditions, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 Organization

6.2.1 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 the 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. The organization chart and the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR. The functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions shall be documented in procedures.

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

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out radiation protection, 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 Unit Staff

The unit staff organization shall include the following:

a. At least two non-licensed operators shall be assigned when the unit is in the power operating condition; and at least one non-licensed operator shall be assigned when the unit is in the hot shutdown, cold shutdown, or refueling conditions. In addition, if the process computer is out of service for greater than 8 hours, at least three non-licensed operators shall be assigned when the unit is in the power operating, hot shutdown, cold shutdown, or refueling conditions.

b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specification 6.2.2.a 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. An individual qualified to implement 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 of on-duty personnel, provided immediate action is taken to fill the required position.

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

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

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

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

e. As a minimum, either the Manager Operations or the General Supervisor Operations shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift supervision 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.

6.3 Unit Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for; the Manager Operations who, in lieu of meeting the senior reactor operator license requirements of ANSI N18.1-1971, shall 1) hold a senior reactor operator license at the time of appointment, or 2) have held a senior reactor operator license at Nine Mile Point Nuclear Station Unit 1 or at a similar unit, or 3) have been certified for equivalent senior reactor operator knowledge; and the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 Procedures

6.4.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and cover the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972;

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

c. Quality assurance for radioactive effluent and radiological environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 6.5.

6.5 Programs and Manuals

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

6.5.1 Offsite Dose Calculation Manual (ODCM)

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

b. The ODCM shall also contain 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 6.6.2 and Specification 6.6.3.

c. Licensee initiated changes to the ODCM:

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

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

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

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

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report 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.

6.5.2 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 Core Spray, Containment Spray, Emergency Cooling, Shutdown Cooling, Reactor Cleanup, Vacuum Relief, Reactor Water Sampling, Containment Atmosphere Dilution (CAD) H_2O_2 Monitor, Drywell Containment Atmosphere Monitoring (CAM), Post Accident Sampling, Radioactive Gaseous Effluent Monitoring (RAGEMS) (the program requirements shall apply to the Post Accident Sampling System and RAGEMS until such time as administrative controls provide for continuous isolation of the associated penetration(s) or a modification eliminates the potential leakage path(s)), Offgas Effluent Stack Monitoring (OGESMS), and Post Accident Vent to Reactor Building Emergency Ventilation. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. System leak test requirements for each system at 24 month intervals.

The provisions of Specification 4.0.1 are applicable to the 24 month frequency for performing system leak test activities.

6.5.3 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 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 from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

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

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

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

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

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

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; and

k. Limitations on venting and purging of the primary containment through the Emergency Ventilation System to maintain releases as low as reasonably achievable.

The provisions of Surveillance Requirement 4.0.1 are applicable to the Radioactive Effluent Controls Program surveillance frequencies.

6.5.4 <u>Inservice Testing Program</u>

This program provides controls for inservice testing of Quality Group A, B, and C pumps and valves.

a. Inservice testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with requirements for American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components specified in Section XI of the applicable ASME Boiler and Pressure Vessel Code Edition and Addenda, subject to the applicable provisions of 10CFR50.55a;

b. The provisions of Specification 4.0.1 are applicable to the normal and accelerated testing frequencies for performing inservice testing activities;

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

6.5.5 <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment 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); and

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is ≤10 Ci, excluding tritium and dissolved or entrained noble gases.

The provisions of Surveillance Requirement 4.0.1 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

6.5.6 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 the Bases without prior NRC approval provided the changes do not involve either of the following:

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

    2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of 6.5.6.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.7 10 CFR 50 Appendix J Testing Program Plan

a. 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. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

   1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topical Report BN-TOP-1, and

   2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

b. The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

c. The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

d. Leakage Rate Surveillance Test acceptance criteria are:

   1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.

   2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.

   3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

   4. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a minimum pathway basis, at all times when containment integrity is required.

e. The provisions of Specification 4.0.1 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

6.6 Reporting Requirements

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

6.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent of >100 mrems and the associated collective deep dose equivalent (reported in man-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ion chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling <20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

6.6.2 Annual Radiological Environmental Operating Report*

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.

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

---

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

6.6.3 Radioactive Effluent Release Report*

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 the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specifications 3.1.7.a and 3.1.7.e.

2. The K_f core flow adjustment factor for Specification 3.1.7.c.

3. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.1.7.c and 3.1.7.e.

4. The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5. The Power/Flow relationship for Specifications 3.1.7.d and 3.1.7.e.

---

* 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.

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. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, (NRC approved version specified in the COLR).

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 shutdown margin (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

6.6.6 Special Reports

Special reports shall be submitted within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.2.2.(b) (12 months).

b. (Deleted)

c. (Deleted)

d. (Deleted)

e. (Deleted)

f. (Deleted)

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months).

h. Accident Monitoring Instrumentation Report, Specification 3.6.11.a (Table 3.6.11-2, Action 3 or 4) (Within 14 days following the event).

6.7 High Radiation Area

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1. All such door and gate keys shall be maintained under the administrative control of the Station Shift Supervisor - Nuclear, radiation protection manager, or his or her designee.

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

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

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

d. Each individual or group entering such an area shall possess one of the following:

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

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

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

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

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

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

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

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

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK

THIS PAGE INTENTIONALLY BLANK