3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Environmental design criteria for the facilities conform to 10CFR50, Appendix A, General Design Criteria 1, 4, and 23. Compatibility of mechanical and electrical equipment with environmental conditions is provided to fulfill the following design criteria:

a) For normal operation, systems and components required to mitigate the consequences of a design basis accident (DBA) or required for a safe shutdown, are designed to remain functional after exposure to the following environmental conditions:

   1) Design temperatures, pressures, and relative humidity values maintained at the equipment location during normal operating by the heating, ventilating, and cooling systems described in Section 9.4.

   2) Maximum expected integrated radiation exposures, for 40 years at the equipment location during normal operation. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

The environmental conditions expected during normal operation are given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. The environmental conditions identified in Table 3.11-1 are for the turbine building and for all elevations in the control building except elevation 806'. The environmental conditions in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are for primary containment, the reactor building and elevation 806' of the control structure.

b) In addition to the normal operation environmental requirements listed in a) above, the systems and components required to mitigate the consequences of a DBA, or to effect a safe shutdown of the reactor are designed to remain functional after exposure to the applicable accident environmental conditions. The applicable environmental conditions are those anticipated to follow the DBA that the systems or components are intended to mitigate and are listed below:

   1) Components Inside Containment

      The temperature, pressure, and humidity inside containment after a design basis Loss of Coolant Accident (LOCA) conditions are indicated in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Containment zones are shown on Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The post-LOCA radiation environment is calculated assuming that 100 percent of the core noble gas inventory, 50 percent of the core halogen inventory, and 1 percent of the core solid fission product inventory are released. The calculational method is in accordance with NUREG 0588, Rev. 1, Section 1.4 and appendix D. The total calculated post-accident dose is the integrated dose from the time of the LOCA to 180 days.

2) Components Outside Containment

The expected temperature, pressure and humidity conditions are specified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

In computing the expected integrated accident doses for equipment in contact with or in proximity to Emergency Core Cooling System (ECCS) - water, it is assumed that 50 percent of the core halogen inventory and 1 percent of the core solid fission product inventory are diluted by the Reactor Coolant System water plus the suppression pool water after a design basis LOCA. For equipment located remotely from ECCS water, the appropriate accidental release is assumed.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Class 1E safety-related equipment is installed in accordance with mechanical and electrical separation requirements and designed and qualified in accordance with the provisions of IEEE 323-1971 and 1974, with appropriate margins, to function properly in the environments listed.

An Environmental Qualification Master Equipment List (EQMEL) is maintained for the SSES equipment which requires environmental qualification through the current Procedure.

1) required to detect a steam or water line accident condition;
2) required to perform a steamline isolation function;
3) required to perform a water line isolation function and could be subjected to the steam environment such as electrical cable or valve operator;
4) required for safety system operation and is located so a steamline break in some other system exposes the safety system equipment to the local accident environment; and,
5) required to track the post-accident environment condition such as pressure, temperature and radiation monitors.

Electrical switchgear and motor control centers required for safety system operation are located outside of the drywell accident environment to ensure operation.

Harsh environments may arise in primary and secondary containments as a result of a Loss of Coolant Accident (LOCA) inside primary containment. In addition, harsh environments may arise in localized areas outside primary containment as a result of a High Energy Line Break (HELB). The environmental conditions (both normal and maximum) to which Class 1E equipment is exposed are given in Table 3.11-1 and Dwgs.
C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4,
C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8,
C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The nonseismic vibration of safety-related equipment conforms to requirements of the following standards:

Equipment	Standard
Diesel Fuel Oil Transfer Pumps	Hydraulics Institute Standards
RHR Service Water Pumps	Hydraulics Institute Standards
Emergency Service Water Pumps	Hydraulics Institute Standards
Control Structure Chilled Water Pumps	Hydraulics Institute Standards
All other Safety-Related Pumps	API 610, Section IV or better
HVAC Fans	ASHRAE Systems Handbook 1973 Edition, Chapter 35, Page 24
Diesel Generator Engines	DEMA Standard Practices for low and medium speed stationary diesel and gas engines
Electric Motors	NEMA MGI

The absence of any significant nonseismic vibration caused by pipe vibration interaction with above equipment is verified in Subsection 3.9.2.1.

A containment spray system may be utilized following the LOCA; therefore, exposed safety-related systems located in the containment are designed to withstand the effects of the containment spray.

3.11.2 QUAIIFICATION TEST AND ANAIYSIS

The qualification tests conform to the requirements of Appendix B, Section XI of 10CFR50.

For the Class 1E equipment, qualification tests and analysis performed on electrical equipment, including motors, are maintained in an auditable manner as discussed in Section 3.11.3. Class 1E equipment installed at SSES is subject at a minimum to the requirements of NUREG-0588, Category II as detailed in Section 3.11.2a.1. Certain new or replacement Class 1E equipment installed at SSES is subject at a minimum to the requirements of IEEE Standard 323-1974 as detailed in Section 3.11.2a.2.

3.11.2.1 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1971

The original Class 1E equipment at SSES have been qualified at a minimum to NUREG-0588, Category II and IEEE Standard 323-1971. This is because SSES received its construction permit before July 1, 1974. Nuclear power plants which received their construction permits after

July 1, 1974 are required to qualify the Class 1E equipment to NUREG-0588, Category I and IEEE Standard 323-1974. New or replacement equipment purchased after May 23, 1980 is qualified to the requirements of 10CFR50.49 (NUREG-0588, Category I; IEEE Standard 323-1974) except in cases where sound reasons to the contrary have been established per the guidelines of Regulatory Guide 1.89.

3.11.2.2 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1974

A large number of pieces of Class 1E equipment at SSES are qualified to NUREG-0588, Category I and to IEEE Standard 323-1974. Qualification to IEEE Standard 323-1974 requires type testing of a prototype to demonstrate that the equipment will perform its safety function in the combined temperature, pressure, humidity, chemical and radiation environment.

Equipment qualified to IEEE 323-1974 is qualified to the test sequence and margins specified in IEEE 323-1974 unless justification for using another sequence or other margins is provided in the SSES EQ documentation.

3.11.2.3 Class 1E Component Environment Design and Qualification for Normal Operation

Class 1E equipment is designed for 40 years of continuous operation in the most severe temperature, pressure, humidity, and radiation environments that exist at the equipment location during normal operation, assuming that proper routine preventive maintenance is performed, such as periodic replacement of seals, packing, and consumable materials. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 provide the maximum normal and maximum DBE values for the environmental parameters of temperature, pressure, and humidity for each area in which Class 1E safety-related equipment is located, as well as the exposures to radiation.

For most equipment, special qualification tests to verify operability at normal operating temperature, pressure, and humidity conditions are not required. Verification for this equipment is based on proven operating capability in similar environments in industrial and previous nuclear power plant applications. The pre-operational and post-operational test programs for safety-related components further ensure that safety-related components will be available when required. Since the normal and accident integrated radiation doses have cumulative effects, the integrated radiation dose during normal operation is discussed in Subsection 3.11.5.3.

3.11.2.4 Class 1E Component Environmental Design and Qualification for Operation After a Design Basis Accident

Class 1E safety-related equipment is designed to remain functional in the most severe combination of temperature, pressure and humidity conditions that exist at the equipment location after a design basis LOCA. This equipment is also designed for the maximum calculated integrated radiation exposure of the LOCA or of the accident, as discussed in Subsection 3.11.5. The temperature, pressure, and humidity environment inside the primary containment after a LOCA is presented and discussed in detail in Subsection 6.2.1. The integrated post accident radiation dose for plant locations in which the equipment is located is given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. In addition, possible steam and feedwater line breaks outside the containment are analytically checked to ensure that no additional qualifications need be applied to components that could be affected by these breaks.

The requirements of the general design criteria, 1, 4, and 23 of Appendix A to 10CFR50, are met as discussed in Section 3.1.

The recommendations contained in the regulatory guides listed below (listings a) through g)) have been utilized as described in Section 3.13. Additional discussion is included in listings f) through j).

a) Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment.

b) Regulatory Guide 1.40, Qualification Test of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants.

   Continuous duty motors used inside the containment are type tested under simulated LOCA conditions. IEEE Standard 334-1971 is used. Insofar as practicable, auxiliary equipment that is part of the installed motor assembly is likewise qualified in accordance with IEEE Standard 334 under simulated design bases event conditions.

c) Regulatory Guide 1.63, Electrical Penetrations Assemblies in Containment Structures for Water Cooled Nuclear Plants.

   Electrical containment penetrations are tested in accordance with IEEE Standard 317-1972. Refer to Section 8.1 for discussion on this guide.

d) Regulatory Guide 1.73, Qualification Test of Electric Valve Operator, Installed Inside the Containment of Nuclear Power Plants.

   Motor operated valves used inside the containment are type tested as a minimum in accordance with IEEE 382-1972 (ANSI N41.6).

e) Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants.

The qualification methods and documentation requirements of IEEE 323-1971 are discussed in Section 3.11.

f) Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, Rev. 2, December 1980.

g) Regulatory Guide 1.131, Qualification Tests of Electric Cables, Filed Splices, and Connections for Light-Water-Cooled Nuclear Power Plants, August 1977.

h) Type tests to ensure acceptability for use in the containment post accident environment are performed for each type of cable in accordance with IEEE Standard 383-1974.

i) Pressure boundary components inside the containment are designed for the temperature, pressure, and humidity environment in accordance with the applicable codes to which the component is constructed. Qualification testing is not considered necessary for such components.

j) A total (normal plus accident) integrated dose of less than 10^4 rad will not affect the strength or properties of materials used; hence, further qualification analyses and tests for components which will be exposed to less than 10^4 rad are not necessary. However, certain electronic equipment such as metal oxide semi-conductive devices are sensitive to radiation levels of less than 10^4. Therefore, radiation qualification is evaluated on a case by case basis even when the postulated accident dose is less than 10^4. For higher integrated doses, components are qualified either by qualification testing or by evaluation of materials used. Reliable accumulated data on radiation effects, such as contained in Reference 2, is used to analyze the dose effects of particular materials.

k) The sources used in calculating radiation levels following LOCA are consistent with those set forth in NUREG-0588, Rev. 1. All the active non-NSSS safety-related equipment located inside the primary containment is designed to withstand the maximum integrated doses during the life of the plant. Suitability of materials used is verified by test for all electrical penetration assembly materials, for an integrated dose rate of at least 5×10^7 rad, in accordance with requirements of IEEE 317-1972.

l) The materials used in the fabrication of the reactor coolant system pressure boundary and other mechanical and structural components are selected to minimize corrosion and hydrogen generation.

3.11.3 QUALIFICATION TEST RESULTS

Environmental qualification documentation for Class 1E electrical equipment is contained in EQ binders maintained by Susquehanna Nuclear Records in File R34. In some cases, additional details of test results for GE safety-related equipment are maintained in a permanent file by GE and can be readily audited. In all cases, the equipment used in Class 1E applications passed the prescribed tests. Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12, identify the temperature pressure, humidity, and radiation environments to which the Class 1E equipment has been qualified.

3.11.4 LOSS OF VENTILATION

The maximum temperatures considered in the sizing of air conditioning systems serving safety-related systems are determined by additive analysis of the following factors:

a) Maximum outdoor design temperature for the geographical area of the plant (both wet bulb and dry bulb readings)

b) Maximum internal piping thermal loads, if applicable, for the room, using maximum normal operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation.

c) Maximum internal electrical load assuming full lighting for the room, and using, if applicable, the maximum control and equipment resistance losses for each mode of operation.

d) Maximum heat transfer from miscellaneous equipment surfaces, if applicable (e.g., outer surface of the diesel generator).

e) Maximum heat transfer from the surface of open pools and tanks, if applicable, using the maximum operating temperature of the contents.

f) Maximum heat transfer from the room envelope including walls, floor, and ceiling, or roof (this value may be negative).

Seismic Category I air conditioning and air cooling systems, described in Section 9.4 are powered from the Class 1E electrical power supplies and are provided for the locations listed in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

These Category I systems are designed so that the single failure of an active mechanical component, or an active or passive electrical component, after a DBA, cannot impair the ability of the systems served by the air conditioning equipment to fulfill their safety functions. Should a train in a Seismic Category I air conditioning system become inoperative during normal operation, sufficient equipment is still available to mitigate the consequences of a DBA.

Two redundant Seismic Category I emergency air conditioning trains are provided for the control room.

Power cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. The allowable current carrying capacity of the cable is based on not exceeding this insulation design temperature while the surrounding air is at an ambient temperature of 150°F (65.5°C) inside the containment and for the rest of the plant area a temperature of 122°F (50°C) or 104°F (40°C) depending on location. The cable ampacity is determined as discussed in Section 8.3.3.1.

Instrumentation cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are environmentally qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. Operating currents of these cables are low and will not cause this temperature to be exceeded at maximum design ambient temperature. Instruments required to operate following a DBA are not located in pipe tunnels, but are mounted outside the tunnels.

3.11.5 ESTIMATED CHEMICAL, PHYSICAL, AND RADIATION ENVIRONMENT

3.11.5.1 Suppression Pool, Residual Heat Removal System, and Emergency Core Cooling System Water Quality

The water in these systems shall not be chemically inhibited. The maximum limits for the suppression pool have been established to be compatible with the primary coolant limits for the shutdown condition and are listed in Table 3.11-7 for comparison.

Observations made of suppression pool water quality over a period of several years in suppression pool with and without coatings, indicate that the feed and bleed to radwaste that occurs during normal system testing and level adjustments maintain the water quality well within the above limits.

During reactor shutdown cooling, the RHR system water mixes with reactor water. Therefore, to insure reactor water quality, as much as practicable of the shutdown cooling piping and equipment shall be flushed with either reactor water or water of the quality specified above for maximum limit.

3.11.5.2 Physical Environment

Engineered safety feature (ESF) systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in Subsection 3.11.2, and Sections 6.2 and 6.3.

The containment atmosphere is maintained below 4 percent by volume hydrogen consistent with the recommendations of Regulatory Guide 1.7 as discussed in Subsection 6.2.5.

3.11.5.3 Radiation Environment

ESF systems and components are designed to perform their safety-related functions after the normal operational exposure plus an accident exposure. The normal operational exposure is based on the design source terms presented in Chapter 11 and Subsection 12.2.1 and the equipment and shielding configurations presented in Section 12.3.

Post-accident ESF system and component radiation exposures are dependent on equipment location. In the containment and control room area, exposures are due to a hypothesized LOCA. Source terms and other accident parameters are consistent with the recommendations of NUREG 0588, Rev. 1.

In the reactor building, post-accident exposures to ECCS systems recirculating depressurized reactor fluids are based on source terms assuming an inventory of 50 per cent of the core halogens and one percent of the core solid fission products. Post Accident exposures to the High-Pressure Coolant Injection and Reactor Core Isolating Cooling Systems are based on steam source terms of control rod drop accident as described in Section 15.4.9. The minimum exposure within the Reactor Building is based on airborne sources originating from one per cent per day drywell leakage of post-accident airborne activity.

Normal, accident, and design (normal plus accident) radiation exposures for plant areas, based on the above assumptions, are presented in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Organic materials that exist within the containment are identified in Subsection 6.1.2. The design radiation exposures identified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are based on gamma radiation exposure only except as noted. The attenuation of beta radiation by small amounts of shielding, such as conduits and jackets for cable and casings for equipment, is evaluated. Where such shielding is not completely effective, the equipment is qualified for radiation exposures which include the appropriate portion of the postulated beta TID.

3.11.6 REFERENCES

3.11-1. J.J. DiNunno, R.E. Baker, F.D. Anderson, and R.L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C. (1962).

3.11-2. J.F. Kircher and R.E. Bowman, "Effects of Radiation on Materials and Components", Van Nostrand Reinhold, New York, 1964.

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

ABEA	KEY (3)	PRESSURE	NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM			
			TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
Control Room	CS1	+.125" wg	80 / 70	55 / 45	.0005	$1.8X10^{2}$	+.125" wg	80	55	<1.0	$1.8X10^{2}$
Cable Spreading Rooms & HVAC Equipment Room, Relay Rooms Elect. Equip. Rooms	CS2	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^{2}$	+.125" wg	90	60	<1.0	$1.8X10^{2}$
Battery Room	CS5	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^{2}$	+.125" wg	80	60	<1.0	$1.8X10^{2}$
Computer Room	CS3	+.125" wg	85 / 65	60 / 40	.0005	$1.8X10^{2}$	+.125" wg	85	60	<1.0	$1.8X10^{2}$
Diesel Generator 'A-D' Rooms (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^{2}$	Atmos	120	50	<1.0	$1.8X10^{2}$
ESW Pumphouse	SW	Atmos	104 / 40	100 / 5	.0005	$1.8X10^{2}$	Atmos	104	100	<1.0	$1.8X10^{2}$
UPS Rooms	CS3	+.125" wg	104 / 65	60 / 40	.0005	$1.8X10^{2}$	+.125" wg	104	60	<1.0	$1.8X10^{2}$
Turbine Building Operating Floor	T2a	Atmos	104	90 / 10	.0025	$8.8X10^{2}$	-.125" wg	104	90	≤1.0	$8.8X10^{2}$
Diesel Generator E Building (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^{2}$	Atmos	120	50	≤1.0	$1.8X10^{2}$
Turbine Building General Areas (Shielded)	T1	-.125" wg	104	90 / 10	.0025	$8.8X10^{2}$	-.125" wg	104	100	≤1.0	$8.8X10^{2}$
HP Turbine	T2b	-.125" wg	-	-	.5	$1.8X10^{4}$	-.125" wg	-	-	≤1.0	$1.8X10^{3}$

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

			NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM			
ABEA	KEY (3)	PRESSURE	TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
LP Turbine	T2c	-.125" wg	-	-	.1	$3.5X10^4$	-.125" wg	-	-	≤1.0	$3.5X10^4$
Feedwater Heaters Condensers	T3	-.125" wg	120	90 / 10	5	$1.8X10^6$	-.125" wg	120	100	≤1.0	$1.8X10^6$
Steam Jet Air Ejectors	T4	-.125" wg	120	90 / 10	15	$5.3X10^6$	-.125" wg	120	100	≤1.0	$5.3X10^6$
Condensate Treatment	T5	-.125" wg	120	90 / 10	10	$3.5X10^6$	-.125" wg	120	100	≤1.0	$3.5X10^6$

(1) Includes integrated accident and normal dose TID for 180 days. After 180 days, the TID is essentially saturated and will not increase significantly.
(2) Pressure, temperature, and humidity maximum are not simultaneous. Above normal pressure, temperature and humidity are considered to persist for 100 days. After 100 days, the thermal environment will be equal to or less than the "maximum" given for normal operation.
(3) Key letter and number identifies a particular group of environmental parameters.
(4) If not otherwise noted, dose is gamma.
(5) For DG rooms: Normal operation means DG in Standby, maximum condition means DG operating.
(6) Maximum Condition Dose rates and TIDs are maximum contact doses in each room, and specific equipment may be subject to a reduced dose based upon the appropriate attenuation factors.
(7) For Beta Sensitive equipment only, the post-accident airborne Beta doses shown in this note must be corrected with appropriate attenuation factors and then added to the tabulated gamma doses to determine total TID.

Area	Max Beta Dose Rate (R/HR)	Beta TID (RAD)
Control Building	2.0	$1.0X10^2$
Turbine Building	20.0	$1.0X10^3$

TABLE 3.11-7

WATER QUALITY

PARAMETER	REACTOR WATER LIMITS SHUTDOWN CONDITION	PRESSURE SUPPRESSION POOL WATER QUALITY EXPECTED	SUPPRESSION POOL WATER MAXIMUM LIMIT
Conductivity	≤ 10 μs/cm at 25°C	≤ 5 μs/cm at 25°C	≤ 10 μs/cm at 25°C
Chlorides (as C1)	≤ 0.5 ppm	≤ 0.2 ppm	≤ 0.5 ppm
pH	5.3 to 8.6 at 25°C	6.0 to 7.5 at 25°C	5.3 to 8.6 at 25°C

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-1 replaced by dwg.
C-1815, Sh. 1

FIGURE 3.11-1, Rev. 56

AutoCAD Figure 3_11_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-2 replaced by dwg. C-1815, Sh. 2 FIGURE 3.11-2, Rev. 51

AutoCAD Figure 3_11_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 3

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-3 replaced by dwg.
C-1815, Sh. 3

FIGURE 3.11-3, Rev. 51

AutoCAD Figure 3_11_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 4

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-4 replaced by dwg. C-1815, Sh. 4 FIGURE 3.11-4, Rev. 56

AutoCAD Figure 3_11_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 5

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-5 replaced by dwg. C-1815, Sh. 5 FIGURE 3.11-5, Rev. 56

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 6

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Figure 3.11-6 replaced by dwg. C-1815, Sh. 6 FIGURE 3.11-6, Rev. 56

AutoCAD Figure 3_11_6.doc

THIS FIGURE HAS BEEN
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Figure 3.11-7 replaced by dwg.
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FIGURE 3.11-7, Rev. 56

AutoCAD Figure 3_11_7.doc

THIS FIGURE HAS BEEN
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Figure 3.11-8 replaced by dwg. C-1815, Sh. 8 FIGURE 3.11-8, Rev. 56

AutoCAD Figure 3_11_8.doc

THIS FIGURE HAS BEEN
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Figure 3.11-9 replaced by dwg.
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FIGURE 3.11-9, Rev. 56

AutoCAD Figure 3_11_9.doc

THIS FIGURE HAS BEEN
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Figure 3.11-10 replaced by dwg.
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FIGURE 3.11-10, Rev. 58

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THIS FIGURE HAS BEEN
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C-1815, Sh. 11

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Figure 3.11-11 replaced by dwg. C-1815, Sh. 11 FIGURE 3.11-11, Rev. 56

AutoCAD Figure 3_11_11.doc

THIS FIGURE HAS BEEN
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Figure 3.11-12 replaced by dwg. C-1815, Sh. 12 FIGURE 3.11-12, Rev. 56

AutoCAD Figure 3_11_12.doc

Appendix 3.11A has been deleted.