1AA Plant Shielding to Provide Access to Vital Areas and Protective Safety Equipment for Post-Accident Operation [II.B.2]

1AA.1 Introduction

General Electric has performed a review of the ABWR Standard Plant post-accident environment in response to NUREG-0737 Item II.B.2. This attachment discusses the results of that review.

1AA.2 Summary of Shielding Design Review

Several alternatives are potentially available to the designer to assure continued equipment availability and performance under post-accident conditions. One is to provide redundant systems and/or components which are qualified to operate in the expected environment. Another is to provide operator access to conduct the operations and to maintain the equipment. This latter alternative would generally be accompanied by appropriate shielding and in many cases would be difficult if not impossible to carry out.

GE has taken the first approach and furthermore has designed the plant so that most responses to transient conditions are automatic, including achieving and maintaining safe-shutdown conditions. The design basis for the ABWR Standard Plant is to require safety-related equipment to be appropriately environmentally qualified and operable from the control room. As a result of this design philosophy and as shown by this review, no changes are necessary to assure that personnel access is adequate or that safety equipment is not degraded because of post-accident operation.

As part of the design of the ABWR Standard Plant, it was necessary to establish the environmental conditions for qualification of safety-related equipment. A result of this design work was an environmental requirement establishing the integrated dose that the equipment must be able to withstand. These values are listed in Appendix 3I.

Another aspect of the review was the manner in which the safety-related equipment is arranged and operated during normal and abnormal operation and postulated accidents. The essence of the ABWR Standard Plant is to achieve and maintain a safe shutdown condition for all postulated accident conditions with all operator actions being conducted from outside the primary and secondary containment zones, principally from the control room.

The purpose of this review is first to verify that, where equipment access is required, it is reasonably accessible outside the primary and secondary containment zones. Secondly, the review should verify that inaccessible equipment is environmentally qualified and is operable from the control room.

The results of the review are:

(1) The period of interest begins with the plant in a safe shutdown condition. Thus, the various safety-related systems needed to achieve safe shutdown conditions have performed as expected, and only the engineered safety features systems (Chapter 6) and auxiliaries, as described later, are required to maintain this condition.

(2) Based upon the accident source terms of Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 including normal operations, the vital equipment exposures will be enveloped based upon the table below:

Area	Gamma (Gy)	Beta (Gy)
Primary Containment	2×10^6	2×10^7
ECCS Rooms	6×10^5	8×10^7
SGTS Area	3×10^7	3×10^2

Each actual area will be environmentally qualified to the area specific envelope as defined in Tables 3I.3-9 through 3I.3-13 and 3I.3-19 through 3I.3-20.

(3) It is not necessary for operating personnel to have access to any place other than the control room, technical support center, post-accident sampling station, sample analysis area, and safety-related nitrogen supply bottles to operate the equipment of interest during the 100-day period. The control room, technical center and sample analysis area are designed to be accessible post-accident. The latter areas are considered accessible on a controlled exposure basis. Direct shine from the containment is less than 3.87E-06 c/kgh in the control room, technical support center, and counting facility and less than 1.29E-04 c/kgh in other vital areas in the Reactor Building.

(4) Access to radwaste is not required, but the Radwaste Building is accessible, since primary containment sump discharges are isolated and secondary containment sump pump power is shed at the onset of the accident. Thus, fission products are not transported to radwaste. The combustible gas control system is operated from the control room; the ABWR does not have a containment isolation reset control area or a manual ECCS alignment area. These functions are provided in the control room.

(5) Following an accident, access is available to electrical equipment rooms containing motor control centers and corridors in the upper Reactor Building (Subsection 12.3.6). This is based on radiation shine from the ECCS rooms and primary containment. While not necessary to maintain safe shutdown, such access can be useful in extending system functionality and plant recovery.

(6) The emergency power supplies (diesel generators) are accessible. However, access is not necessary, since the equipment is environmentally qualified.

1AA.3 Containment Description and Post Accident Operations

1AA.3.1 Description of Primary/Secondary Containment

The ABWR design includes many features to assure that personnel occupancy is not unduly limited and safety equipment is not degraded by post-accident radiation fields. These features are detailed in Tier 2 and only a brief summary description and Tier 2 reference are provided here for emphasis.

The configuration of the pressure suppression primary containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the primary containment atmosphere following an accident is thereby substantially reduced compared to the Regulatory Guide 1.3 source terms.

Primary containment leakage is limited to less than one half percent of the primary containment atmosphere per day. The surrounding secondary containment is kept at a negative pressure with respect to the environment permitting monitoring and treating all radioactive leakage from the primary containment.

The Standby Gas Treatment System (SGTS) operates automatically from the beginning of an accident to control the secondary containment pressure to -6.4 mm w.g. The large volume of this portion of the Reactor Building acts as a mixing chamber to dilute any primary containment leakage before processing by the SGTS and discharge to the environment. Discharge of radioactivity is thus controlled and reduced. Radioactivity content of secondary containment atmosphere is reduced with time by SGTS treatment as well as by decay. (However, prior removal of halogens by scrubbing in the suppression pool offsets the necessity of this treatment.)

Each ECCS pump and supporting equipment is located in an individual shielded, watertight compartment. Spread of radioactivity among compartments is thus limited. Radiation to the other equipment areas and corridors of the Reactor Building is limited to shine through the walls; there is no airborne radiation in these other areas. As these become accessible after an accident, any component failures can be repaired, thereby improving systems availability.

1AA.3.2 Vital Area and Systems

A vital area is any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas which must be considered as vital after an accident are the control room, technical support center, sampling station, sample analysis area and the HPIN nitrogen supply bottles.

The vital areas also include consideration (in accordance with NUREG- 0737, II.B.2) of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control center and radwaste control panels. However, the ABWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room. Those vital areas which are normally areas of mild environment, allowing unlimited access, are not reviewed for access.

Essential systems specific to the ABWR to be considered post-accident are those for the ECCS, fission product control and the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1AA.3.3 Post Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10CFR100 guidelines.

Many of the safety-related systems are required for reactor protection or to achieve a safe shutdown condition. However, they are not necessarily needed once a safe shutdown condition is achieved. Thus, the systems considered herein are the engineered safety features (ESF) (Chapter 6) used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the post-accident sampling station, the sample analysis area, and two manual nitrogen reserve supply valves.

1AA.4 Design Review Bases

1AA.4.1 Radioactive Source Term and Dose Rates

The radioactive source term used is equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas. There is no leakage outside of secondary containment other than via the SGTS.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <0.15 mSv/h.

Design dose rates for personnel in a vital area are such that the guidelines of General Design Criteria (GDC) 19 (i.e., <0.05 Sv whole body or its equivalent to any part of the body) are not exceeded for the duration of the accident, based upon expected occupancy and protection.

1AA.4.2 Accidents Used as the Basis for the Specified Radioactivity Release

Table 15.0-3 summarizes the various design basis accidents and associated potential for fuel rod failure. This chapter also provides the accident parameters. Of those accidents, only the DBA-LOCA may produce 100% failed fuel rods under NRC worst-case assumptions. The rod drop accident and fuel handling accident are the only other accidents postulated as leading to failed fuel rods with the potential consequence of radioactivity releases.

For the fuel handling accident, the reactor is either shutdown and cooled or is operating normally if the accident is in the spent fuel storage pool. Based on the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the reactor building (Table 15.7-9) is released to the environment over a 2-hour period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-10 and the calculated exposure in Table 15.7-11. The exposures are within the guidelines of 10CFR100. Thus, recovery is possible well within the specified 100-day equipment qualification period. ECCS equipment is not affected by this accident and radiation in the ECCS area is not increased. This accident is not considered further.

The postulated control rod drop accident (Subsection 15.4.9) is one which occurs without a pipe break and so may require depressurization to attain long-term core cooling with the RHR System. Normally, this accident is terminated by a scram, and the plant is cooled and recovers. The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident. This accident is not further considered.

The DBA-LOCA is the accident producing the limiting conditions of interest for this design review. In this accident the reactor is depressurized and reactor water mixes with suppression pool water in the process of keeping the fuel covered and cooled. Fission products are assumed to be essentially instantaneously released and mixed in the containment atmospheres and suppression pool-reactor water volumes.

1AA.4.3 Availability of Offsite Power

The availability of offsite power is not influenced by plant accident conditions. Loss of offsite power may be assumed as occurring coincident with the beginning of the accident sequence; however, continued absence of offsite power for the accident duration is not realistic. While restoration of offsite power is not a necessary condition for maintaining core cooling, its availability can permit operation of other plant systems which would not otherwise be permitted by emergency power restrictions (e.g., operation of the pneumatic air system, non-safety-related HVAC systems and other systems useful to plant cleanup and recovery).

Based on Table 19D.3-3, the probability for offsite power recovery is estimated to be very high in 8 hours. This is conservative, since the longest time for restoration of offsite power was six hours for the Pennsylvania-New Jersey-Maryland interconnection. The grid used as a basis for the probabilistic risk assessment is presented in Subsection 15D.3.

1AA.4.4 Radiation Qualification Conditions

The safety-related equipment requiring review for qualification is only that necessary for post-accident operations and for providing information for assuring post-accident control.

In 10CFR50, the long-term cooling capability is given as follows: "...decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core." A 100-day period has been selected as a sufficient extended period permitting site and facility response to terminate the event.

As part of the design review process, a set of reference conditions is necessary for comparing expected post-accident radiation exposures. Appendix 3I defines the environmental conditions for safety-related equipment zones for periods of 60 years normal operations, including anticipated tests and abnormal events, and six months following the DBA-LOCA. These conditions are upper bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. In effect, these are specification values, and equipment will be qualified to meet or exceed these values.

Radiation sources in the secondary containment (especially the ECCS rooms of the Reactor Building) are the same as the Table 1AA-1 design basis values for water sources. For airborne radiation sources, the plant design basis of Table 1AA-1 for air is used. Primary containment leakage is assumed to occur in each of the individual secondary containment compartments. This leakage is limited by the fission product control systems (Subsection 6.5.3). As previously noted, no credit has been taken for the radio-halogen scrubbing, which is an inherent feature of the BWR.

1AA.5 Results of the Review

1AA.5.1 Systems Required Post-Accident

This section establishes the various systems equipment which are required to function following an accident along with their locations. The expected habitability conditions and access and control needs are identified for the required post-accident period.

1AA.5.1.1 Necessary Post-Accident Functions and Systems

Following an accident and assuming that immediate plant recovery is not possible, the following functions* are necessary:

(1) Reactivity control

(2) Reactor core cooling

(3) Reactor coolant system integrity

(4) Primary reactor containment integrity

(5) Radioactive effluent control

Reactivity control is a short-term function and is achieved when the reactor is shutdown. The remaining functions are achieved in the longer term post-accident period by use of:

(a) The Emergency Core Cooling System (ECCS) and their auxiliaries (for reactor core cooling)

(b) The Combustible Gas Control System (CGCS) and auxiliaries (for primary containment and reactor coolant system integrity)

(c) The fission product removal and control system and auxiliaries (for radioactive effluent control)

(d) Instrumentation and controls and power for accident monitoring and functioning of the necessary systems and associated habitability systems

* ANSI/ANS 4.5 Criteria for Accident Monitoring Functions in Light Water Reactors.

Tables 1AA-2 through 1AA-5 are generated to show:

(i) What major equipment and systems are required to function and thereby define the systems for review

(ii) The redundant equipment locations by divisional isolated room or area and containment or building

1AA.5.1.2 Emergency Core Cooling Systems and Auxiliaries

Table 1AA-2 lists various systems related to cooling the fuel under post-accident conditions as described in Section 6.3 and Subsection 9.4.5.2 (HVAC). This table shows ECCS equipment and equipment coolers in an ECCS room. Instrumentation transmitters are in adjoining areas. The required power and cooling water in the same division are described in Subsection 1AA.5.1.5. All perform together to provide an ECCS function.

The Automatic Depressurization System (ADS) function is described in Subsection 1.2.2.2.2.4. A postulated non-break or small break accident could require continued need for the depressurization function until the RHR System is placed in the shutdown reactor cooling mode. In the case of a non-break or a small break accident, the majority of the fission products would be released via the safety/relief valves to the suppression pool and hence to the containment, rather than direct mixing through the supersession pool vents, as would occur following a DBA-LOCA. In either case, the distribution of fission products is assumed to be the same as for the DBA-LOCA even though, realistically, a significant portion of halogens and solid fission products would be retained in the reactor pressure vessel. Thus, the results as they apply to the ADS are very conservative. The pneumatic nitrogen supply for the ADS and other containment valves is included in Table 1AA-3 as a portion of the combustible gas control. The hand-operated nitrogen reserve supply valves P54-F017C and D are accessible outside the secondary containment, if needed, to mitigate a large leak.

The high pressure core flooder (HPCF) and the low pressure flooder loop (LPFL) functions are described in Subsections 1.2.2.5.2 and 1.2.2.5.1.1, respectively. The cooling function can also satisfy the containment cooling function in that, by cooling suppression pool water, which is the source of water flowing to the reactor, the containment source of heat is also removed. The wetwell/drywell sprays are described in Subsection 1.2.2.5.1.3.

The fuel pool cooling function (Subsection 1.2.2.7.2) is also included on the basis that a recently unloaded fuel batch could require continued cooling during the post-accident period. This function is also supplemented by the RHR Fuel Pool Cooling Mode as described in Subsection 9.1.3.2. The RHR equipment is environmentally qualified, so access is not required and redundancy is included in system components.

The locations of selected associated valves and instrument transmitters are included. These do not represent all of this type of equipment which is environmentally qualified, safety-related, or included in the systems of Table 3.2-1. It does however, represent principal components which are needed to operate, generally during post accident operations. For example, most

ECCS valves are normally open, and only a pump discharge valve needs to open to direct water to the reactor. Similarly, the instrument transmitters shown are those which would provide information on long-term system performance post-accident. Control room instrumentation is not listed, since it is all in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under main steam (B21) are those for ECCS function or monitoring reactor vessel level. Suppression pool level is included with the HPCF instrumentation.

1AA.5.1.3 Combustible Gas Control Systems and Auxiliaries

Flammability control in the primary containment is achieved by an inert atmosphere during all plant operating modes except for operator access during refueling and maintenance. The high pressure nitrogen (HPIN) gas supply is described in Subsection 1.2.2.12.13. The Containment Atmospheric Monitoring System (CAMS) measures and records containment oxygen/hydrogen concentrations under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.6.1.6). Table 1AA-3 lists the combustible gas control principal components and their locations.

1AA.5.1.4 Fission Product Removal and Control Systems and Auxiliaries

Engineered Safety Feature (ESF) filter systems are the Standby Gas Treatment System (SGTS) and the control building Outdoor Air Cleanup System. Both consist of redundant systems designed for accident conditions and are controlled from the control room. The SGTS filters the gaseous effluent from the primary and secondary containment when required to limit the discharge of radioactivity to the environment. The system function is described in Subsection 1.2.2.15.4.

A portion of the Control Building heating ventilating and air-conditioning (HVAC) provides detection and limits the introduction of radioactive material and smoke into the control room. This portion is described Subsection 9.4.1.1.3.

The CAMS described in the previous section also measures and records containment area radiation under post-accident conditions. A post-accident sampling system (PASS) obtains containment atmosphere and reactor water samples for chemical and radiochemical analysis in the laboratory. Delayed sampling, shielding, remote operated valves and sample transporting casks are utilized to reduce radiation exposure. The samples are manually transported between the PASS room in the Reactor Building and the analysis laboratory in the service building. The system is described in Subsection 9.3.2.3.1. Table 1AA-4 lists the fission product removal control components and locations.

1AA.5.1.5 Instrumentation and Control, Power, and Habitability Systems and Auxiliaries

Most of the post-accident instrumentation and control system equipment is listed with the applicable equipment in Tables 1AA-2, 1AA-3 and 1AA-4. The remaining instrumentation and control equipment is included with the power and habitability systems equipment listed in Table 1AA-5. Instrumentation is consistent with the post accident phase variables monitored by the Post-Accident Monitoring (PAM) System listed in Table 7.5.2.

Standby AC power is supplied by three diesel generators in separate electrical divisions (Subsection 1.2.2.13.13). The diesel generators, switchgear and motor control centers are included in the unit Class 1E AC power system described in Subsection 1.2.2.13.14.1. Storage batteries are the standby power source for the unit Class 1E DC power system described in Subsection 1.2.2.13.12.2. The safety system logic and control power system is described in Subsection 1.2.2.13.14.1.

Habitability systems ensure that the operator can remain in the control room and take appropriate action for post-accident operations. The control building includes all the instrumentation and controls necessary for operating the systems required under post- accident conditions.

The control room, control and reactor building HVAC essential equipment are a portion of the plant environmental control of temperature, pressure, humidity and airborne contamination described in Subsection 1.2.2.16.5 (1), (4), (5), (7) and (8). HVAC units controlling the local room environments are included with respective equipment in Tables 1AA-2, 1AA-3 and 1AA-4. The major HVAC equipment and locations are listed in Table 1AA-5.

The Reactor Building Cooling Water (RBCW) System provides cooling water to designated equipment in the Reactor Building (including containment) as described in Subsection 1.2.2.12.3. The HVAC Emergency Cooling Water (HECW) System provides chilled water to designated equipment in the control building as described in Subsection 1.2.2.12.6.

Table 1AA-1 Radiation Source Comparison

Activity Group	% Core Inventory Released		
	R.G.1.3	R.G.1.7	Plant Design Basis
Air			
Noble Gases	100	100	100*
Halogens	25	—	25*
All Remaining	—	—	—
Water			
Noble Gases	0	—	0
Halogens	—	50	50†
All Remaining	—	1	1†

* Uniformly mixed within the primary containment boundary

† Uniformly mixed in the suppression pool and reactor coolant

Table 1AA-2 Post-Accident Emergency Core Cooling Systems and Auxiliaries

Equipment	MPL	Location
ADS & Transmitters		
SR Valve	B21-F010A,C,F,H,L,N,R,T	Upper Drywell (PC)
SR Accumulator	B21-A004A,C,F,H,L,N,R,T	Upper Drywell (PC)
Rx Water Level (ADS,RHR)	B21-LT003A thru H	Instrument Rack Rm. (SC)
Rx Water Level (HPCF)	B21-LT001A,B,C,D	Instrument Rack Rm. (SC)
Rx Pressure (RHR)	B21-PT301A,B,C,D	Instrument Rack Rm. (SC)
DW Pressure (HPCF, RHR)	B21-PT025A,B,C,D	Instrument Rack Rm. (SC)
HPCF		
Pumps	E22-C001B,C	HPCF Rm. B,C (SC)
SP Suction Valve	E22-F006B,C	HPCF Rm. B,C (SC)
Rx Injection Valve	E22-F003B,C	Valve Rm. B,C (SC)
CST Suction Valve	E22-F001B,C	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D102,106	HPCF Rm. B,C (SC)
CST Water Level	P13-LT001A,B,C,D	HPCF Rm. B,C (SC)
Flow	E22-FT008B1,B2,C1,C2	By HPCF Rm. B,C (SC)
Suction Pressure	E22-PT002,003; B,C	By HPCF Rm. B,C (SC)
Injection Pressure	E22-PT006,007; B,C	By HPCF Rm. B,C (SC)
SP Water Level	T31-LT0058A,B,C,D	By HPCF Rm. B,C (SC)
LPCF		
Pump	E11-C001A,B,C	RHR Rm. A,B,C (SC)
Heat Exchanger	E11-B001A,B,C	RHR Rm. A,B,C (SC)
RCW Discharge Valve	P21-F013A,B,C	RHR Rm. A,B,C (SC)
SP Suction Valve	E11-F001A,B,C	RHR Rm. A,B,C (SC)
Rx Injection Valve	E11-F005A,B,C	Valve Rm. A,B,C (SC)
Rx Return Valve	E11-F010,011,012;A,B,C	Valve Rm. A,B,C (SC)
DW Spray Valve	E11-F017,018;B,C	Valve Rm. B,C (SC)
WW Spray Valve	E11-F019B,C	Valve Rm. B,C (SC)
FPC Supply Valve	E11-F015A,B,C	Valve Rm. A,B,C (SC)
FPS Supply Valve	E11-F101,102,103	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D103,104,105	RHR Rm. (SC)
Flow	E11-FT008A1,B1,C1	By RHR Rm. A,B,C (SC)

Table 1AA-2 Post-Accident Emergency Core Cooling Systems and Auxiliaries (Continued)

Equipment	MPL	Location
Flow	E11-FT008A2,B2,C2	By RHR Rm. A,B,C (SC)
RCW Flow	P21-FT008A,B,C	By RHR Rm. A,B,C (SC)
Hx I/O Temperature	E11-TT006,007;A,B,C	By RHR Rm. A,B,C (SC)
Discharge Pressure	E11-PT004A thru G	By RHR Rm. A,B,C (SC)
DW Temperature	T31-TT/SSA051,053	Inst. Rack Rm. (SC)
DW/WW Pressure Ratio	T31-PT055A,B	Inst. Rack Rm. (SC)
WW Pressure	T31-PT056A,B	Inst. Rack Rm. (SC)
DW Pressure	T31-PT054	Inst. Rack Rm. (SC)
FPCS		
Pump	G41-C001A,B	FPC Pump Rm. (SC)
Heat Exchanger	G41-B001A,B	FPC Hx Rm. (SC)
Pump Discharge Valve	G41-F021A,B	FPC Valve Rm. (SC)
Essential HVH (HVAC)	U41-D107, 108	FPC Valve Rm. (SC)
Flow	G41-FT006A,B	By FPC Pump Rm. (SC)
Suction Pressure	G41-PT003A,B	By FPC Pump Rm. (SC)
Skimmer ST Level	G41-LT020	Refueling Floor(SC)

(PC)—Primary Containment
(SC)—Secondary Containment

Table 1AA-3 Post-Accident Combustible Gas Control Systems and Auxiliaries

Equipment	MPL	Location
HPIN		
Nitrogen Storage Bottles	P54-A001A Thru V	By Valve Rm (RB)
Supply Pressure	P54-PT002A, B, 004, 005	By Valve Rm (RB)
CAMS		
Hydrogen, Oxygen Elements	D23-H_2, O_2 Rack A,B	CAMS Rm. A,B (SC)
Gas Measurement	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
Gas Elements	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
DW Gas Valve	D23-F004A,B	CAMS Rm. A,B (SC)
WW Gas Valve	D23-F006A,B	CAMS Rm. A,B (SC)
Essential HVH (HVAC)	U41-D113,114	CAMS Rm. A,B (SC)
Gas Supply	D23-Gas Cyl. Rack A,B	CAMS Rm. A,B (RB)

(PC)—Primary Containment
(SC)—Secondary Containment
(RB)—Reactor Building outside (Secondary Containment)

Table 1AA-4 Post-Accident Fission Product Removal and Control Systems and Auxiliaries

Equipment	MPL	Location
SGTS		
Exhaust Fan	T22-C001B, C	Fan/Dryer Rm. (SC)
Charcoal Filter	T22-D002C,D001B	Filter Train Rm. (SC)
PC Inlet Valve	T22-F002A,B	Fan/Dryer Rm. (SC)
SC Inlet Valve	T22-F001A,B	Fan/Dryer Rm. (SC)
Stack Outlet Valve	T22-F004A,B	Filter Train Rm. (SC)
PC (DW,WW) Isolation Valves	T31-F004,006,008	Valve Rm. (SC)
Essential HVH (HVAC)	U41-D111,112	SGTS HVH Rm. (SC)
Radiation (Ion/Scint.)	D11-RE002,011;A,B	SGTS Monitor Rm. (RB)
Sampling Rack	H22-P250	By SGTS (SC)
Flow	T22-FT018B,C	By Filter Train. Rm. (RB)
Filter Moisture	T22-MT011B,C & 012B,C	By Filter Train. Rm. (RB)
CR HVAC		
Emerg. Recirculation Fan	U41-C603B	CR HVAC Rm. A,B (CB)
Emerg. Charcoal Filter Unit	U41-B A,B	CR HVAC Rm. A,B (CB)
Air Intake Isolation Valves	U41-F A,B	CR HVAC Rm. A,B (CB)
PASS		
Conditioning/Holding Rack	P91	(SC)
Sampling/Casks Rack	P91	PASS Rack Rm. (RB)
LPCF Supply Valve	E11-F045,046; A	(SC)
DW/WW Gas (CAMS) Valve	D23	(SC)
Control Panel (PT,TT)	H22	PASS Rack Rm. (RB)
Chemical Radiological Analysis		Laboratory (SB)
Stack		
Radiation (Ion/Scint.)	D11-RE041,043; A,B	Stack (RB)
Monitor Racks, Control Rod	H21,H22	Stack Monitoring Rm.(RB)

(CB)—Control Building
(SC)—Secondary Containment
(RB)—Reactor Building outside (Secondary Containment)
(SB)—Service Building

Table 1AA-5 Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries

Equipment	MPL	Location
Instrumentation & Controls		
Post-Accident I&C	H11-Post-Accident	Control & Panel Rms. (CB)
Power		
DC Supply	R42-Storage Batteries	Battery Rm. (CB)
ESF HV&LV Switchgear	R22-Post-Accident	Emerg. Electric Rm. A,B,C (RB)
ESF Motor Control Center	R24-Post-Accident	Emerg. Electric Rm. A,B,C (RB)
Diesel Generator & Auxiliaries	R43-DG A,B,C	DG Rm. A,B,C (RB)
DG Motor Control Center	R43-P001A,B,C	DG MCC Rm. A,B,C (RB)
Supply Fan (HVAC)	U41-C201A,E,204B,F 207C, G	DG Supply Fan Rm. A,B,C (RB)
Exhaust Fan (HVAC)	U41-202A,E,205B,F 207C, G	DG Exhaust Fan Rm. A,B,C (RB)
Essen. Fresh Air Fan (HVAC)	U41-203A,E,206B,F 209C,G	DG Essen. Fan Rm. A,B,C (RB)
RCW Discharge Valve	P21-F055A thru F	DG Rm. A,B,C (RB)
Control Panel (H21)	R43-P002,003C,004;A,B,C	DG Control Pnl. Rm. A,B,C (RB)
CB HVAC		
Supply Fan	U41-C606B,F,608C,G	E/HVAC Rm. A,B,C (CB)
Exhaust Fan	U41-C607B,F,609C,G	E/HVAC Rm. A,B,C (CB)
MCR Supply Fan	U41-C601B,F,604A,E	CR HVAC Rm. A,B (CB)
MCR Exhaust Fan	U41-C602B,F,605A,E	CR HVAC Rm. A,B (CB)
RCW		
Pump	P21-C001A thru G	Pump Rm. A,B,C (CB)
Hx Return Valve	P21-F004D, E, F, A, B, C, D, G, H, J	Hx Rm. A,B,C (CB)
Non-Post-Accident Supply Valve	P21-F074A,B,C	(RB)
Non-Post-Accident Return Valve	P21-F082A,B,C	(RB)
Flow	P21-FT006A,B,C	By Pump Rm. A,B,C (CB)
Pressure	P21-PT004A,B,C	By Pump Rm. A,B,C (CB)
Surge Tank Level	P21-LT013A,B,C	By Surge Tank A,B,C (RB)

Table 1AA-5 Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries (Continued)

Equipment	MPL	Location
HECW		
Pump	P25-C001A,B,C,E,F	Chiller Rm. A,B,C (CB)
Refrigerator	P25-D001A,B,C,E,F	Chiller Rm. A,B,C (CB)
Pressure Control Valve	P25-F012 B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F005 B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F016 A,B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F022 A,B,C	(RB)
RCW Temp. Control Valve	P21-F025 A,B,C,E,F	(CB)
Instrument Air		
Compressor	P52-C001,002	Inst.Air Rm. (RB)

(RB)—Reactor Building outside (Secondary Containment)
(CB)—Control Building