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*Security Notice*
*This letter forwards Security-Related information in accordance with 10CFR2.390. Upon removal of Enclosure 3, the balance of this letter may be considered non-Security-Related.*

MFN 08-227

Docket No. 52-010

May 9, 2008

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information RAI Letter No. 60 Related to the ESBWR Design Certification – Radiation Protection – RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated September 18, 2006. (Reference 1). Initial responses to RAIs 12.4-32 and 12.4-33 were provided in Reference 2. GEH responses to RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01 are addressed in Enclosure 1. Enclosure 2 contains DCD Tier 2 Markups that are not security-related.

Enclosure 3 contains Security-Related DCD Figures identified by the designation "{{{Security-Related Information - Withhold Under 10 CFR 2.390}}}." GEH hereby requests this information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. No public version of these security-related DCD Markups are provided in these RAI responses since they would be blank pages with only figure titles and figure numbers; however, DCD Tier 2, Revision 5 will contain public versions of these figures.

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NRO

If you have any questions or require additional information, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

References:

1. MFN 06-342, Letter from U.S. Nuclear Regulatory Commission to David H. Hinds, GEH, *Request For Additional Information Letter No. 60 Related To ESBWR Design Certification Application*, dated September 18, 2006.
2. MFN 06-477, Letter from David H. Hinds, GEH to U.S. Nuclear Regulatory Commission *Response to Portion of NRC Request for Additional Information Letter No. 60 – Radiation Protection – RAI Numbers 12.3-4, 12.3-5, 12.4-9, 12.4-13, 12.4-18, 12.4-20, 12.4-32 and 12.4-33*, dated November 22, 2006.

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application – Radiation Protection – RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01
2. Response to Portion of NRC Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application – Radiation Protection – RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01– DCD Tier 2 Markups – Non-Security-Related
3. Response to Portion of NRC Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application – Radiation Protection – RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01– DCD Tier 2 Markups – Security-Related Information - Withhold Under 10 CFR 2.390

cc: AE Cubbage USNRC (with enclosure)
GB Stramback GEH/San Jose (with enclosure)
RE Brown GEH/Wilmington (with enclosure)
eDRF 0000-0083-7656R1 and 0000-0085-0437 – RAI 12.4-31
0000-0082-9537 – RAIs 12.4-32S01 and 12.4-33S01

# ENCLOSURE 1

# MFN 08-227

# Response to Portion of NRC Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application

## Radiation Protection

## RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01

**NRC RAI 12.4-31:**

*The post accident radiation zones on DCD Tier 2, Figures 12.3-43 through 12.3-51 are incomplete. Layout drawings are only provided for the "Nuclear Island" and then only the dose rates in the vital areas and "access pathways" are provided. Although the legends on these drawings go up to Zone I (>100 Rem/hr), with the exception of one area on Figure 12.3-51, no area greater than Zone F (1 Rem/hr) is indicated on any of the figures. Provide a complete set of post accident radiation zone drawings. Identify on these drawings the location of: (1) those systems and components that contain post accident materials outside of the primary containment listed under Item II.D.3 of DCD Tier 2, Table 1A-1; (2) each specific area (not just the general room) requiring access to mitigate the consequences of an accident listed under Item II.B.2 of DCD Tier 2, Table 1A-1 (including technical support center and health physics facilities); and (3) the personnel access routes to, and egress routes from, these areas (not just a listing of the general rooms and stairs). Provide a detailed description of personnel actions to be taken in each area, the significant radiation sources associated with each, and an analysis of the radiation "mission" dose received (including dose from access and egress).*

**GEH Response:**

(1) Item II.D.3 of NUREG-0737 refers to "Direct Indication of Relief-and Safety-Valve Position"; GEH believes that the reference should be Item III.D.1.1, as described in the DCD Tier 2 Table 1A-1 "Primary coolant outside containment". In case of an accident, all systems and components that contain post-accident materials are initially contained inside the primary containment and out of any area requiring post-accident access. Dose calculations do not include activity contained in these systems. The systems listed in DCD Tier 2, for NUREG-0737, item III.D.1.1 are located on the accident radiation zone map below (this does not include Reactor Water Cleanup piping evaluated in response to RAI 6.2-140S01, submitted via GEH letter MFN 08-332 dated April 4, 2008):

Fuel and Auxiliary Pools Cooling System (FAPCS) – Figures 12.3-43 and 12.3-44

Isolation Condenser System (ICS) – Figure 12.3-50

Containment Monitoring System (CMS) instrument racks are located at Elevation –11,500 in rooms 1100, 1101, 1102 and 1103 – Figure 12.3-43

Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) – Figures 12.3-43 and 12.3-44

Control Rod Drive System (CRD) – Figures 12.3-43 and 12.3-44

(2) The specific areas analyzed that may be accessed beyond 72 hours and long-term post-accident recovery actions to mitigate the consequences of an accident are:

- Control room

- Technical support center
- Remote shutdown panel rooms
- Standby liquid control pump room
- Health physics facility (counting room)
- IC/PCC Pool Refill Valve
- Electrical Equipment Division 1, 2, 3 and 4
- Nonsafety-related DCIS Rooms
- Diesel generator control rooms
- All access and egress pathways to these rooms

These areas, and their access pathways, are located in the reactor building, control building, electrical building, service building, and outside. No other areas requiring post-accident access were identified in the rest of the buildings.

DCD Tier 2, Figures 12.3-43 through 12.3-51i will be revised to reflect the maximum post-accident dose rates for the buildings that contain areas requiring post-accident access.

New Figures 12.3-74 through 12.3-90 provides the areas requiring post-accident access and personnel access and egress.

The actions beyond 72 hours and actions for long-term post-accident recovery are listed in DCD Tier 2, Table 12.3-11.

For all potential actions to be performed in case of an accident in the areas requiring post-accident access, dose rates along the access pathways and at locations within those areas are calculated. The airborne radiation source term after a LOCA, as indicated in DCD Tier 2 Chapter 15, is taken as input data using the methodology of Regulatory Guide 1.183. The potential direct dose from adjacent rooms has been also considered.

The post-accident dose rates for each time interval at each area requiring post-accident access and pathway are shown in the attached new DCD Tier 2, Tables 12.3-12 and 12.3-13.

(3) The radiation "mission" dose received performing long term recovery actions in the areas requiring post-accident access have been calculated and the results are listed in the new DCD Tier 2 Tables 12.3-14 through 12.3-17. The following calculation criteria have been considered:

- The long-term recovery actions in the areas requiring post-accident access are expected to be required well after 72 hours. However, the mission doses have been conservatively calculated as if the mission occurs at 72 hours.
- The airborne doses, both due to air submersion and inhalation, have been considered.

- Round trip travel, and occupancy times have been taken into account. The mission doses are calculated based on the dose rates of the various rooms.
- A dose reduction factor is assumed by using a pressure demand self-contained breathing apparatus.

The expected post-accident radiation mission doses in the areas requiring post-accident access are shown in new DCD Tier 2 Tables 12.3-14 through 12.3-17, and are below the 0.05 Sv established limit in GDC-19.

Post-accident radiation doses in the control room are also described in DCD Tier 2 Section 15.4, and are also below the 0.05 Sv established limit in GDC-19. The technical support center (TSC) is designed with the same radiological controls as the control room, and accordingly, the expected post-accident radiation doses are also below the 0.05 Sv criterion.

**DCD Impact:**

DCD Tier 2 Subsections 12.3.5, 12.3.6, Table 1A-1, Section 1B, Table 1B-3, and Figures 12.3-43 through 12.3-51i will be revised as shown on the attached markups.

New Tables 12.3-11 through 12.3-17 and Figures 12.3-74 through 12.3-90 will be added in DCD Tier 2 Revision 5 as noted on the attached markups.

**RAI 12.4-32S01 and 12.4-33S01:**

*The last sentence of subsection 12.3.6 is not clear. The post-accident radiation zone maps should be based on the highest expected radiation dose rates under design basis accident conditions, as stated earlier in the subsection. The issues of whether the control room meets GDC 19, and that access to vital areas of the plant during accidents meet NUREG 0737 II.B.2 (50.34(f) (2)(vii)), or that the zone maps support the conclusions, is the subject of RAIs 12.4-31, 33 and 12.3-10. The response to 12.4-32 is incomplete as it refers the answer to RAI answers that have not been submitted.*

**GEH Response:**

DCD Section 12.3.6 will be revised as described in the attached mark-up to the response to RAI 12.4-31.

Access ways to vital areas and the post-accident radiation zone maps, based on the highest expected radiation dose rates under design basis accident conditions, will be included in the next revision of the DCD Tier 2 as described in the attached RAI 12.4-31 response mark-ups.

**DCD Impact:**

No DCD changes will be made in response to this RAI.

# Enclosure 2

# MFN 08-227

# Response to Portion of NRC Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application

## – Radiation Protection

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## – RAI Numbers 12.4-31, 12.4-32S01 and 12.4-33S01

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## – DCD Tier 2 Markups – Non-Security-Related

Process Radiation Monitors, described in Subsection 11.5.3, monitor ventilation paths from the fuel storage area and, in addition to isolating the appropriate ventilation path upon receipt of high radiation, provide indication and alarms to the operator. Area Radiation Monitors, listed in Table 12.3-2, are provided in fuel storage areas to detect high radiation levels and provide visual and audible indication to operating personnel.

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV to 7 MeV. The energy dependence does not exceed 20% of point from 100 keV to 3 MeV. The overall system design accuracy is within 10% of equivalent linear full-scale output for any decade.

The alarm setpoints are established in the field following equipment installation at the site. The exact settings are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures. The radiation alarm setpoint for each channel is set slightly above the background radiation level that is normal in the area where the monitor is located.

Each channel is calibrated based on a pseudo input signal to verify monitor response. Each detector is calibrated using a radioactive source traceable to the National Institute of Standards and Technology. The area radiation monitors are checked and calibrated periodically.

The ARMS is designed to provide early detection and warning for personnel protection to ensure occupational radiation exposures are as low as reasonably achievable (ALARA) in accordance with guidelines stipulated in RG 8.2 (Reference 12.3-14) and RG 8.8. Also, the ARMS include instrumentation in crucial areas of the reactor building where access may be required to service safety-related equipment following a LOCA event.

### 12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the ~~100-day~~ post-accident period are the ~~main~~ control room, the technical support center, the electrical equipment rooms for divisions 1, 2, 3, and 4, the remote shutdown panel, ~~the primary containment sampling locations,~~ the Standby Liquid Control System (SLCS) pump room, the nonsafety-related Distributed Control and Information System (DCIS) rooms, the health physics facility (counting room), ~~the isolation condenser (IC) pool refill nozzles~~the ICC/PCC and fuel pool refill valves, and the ~~nitrogen gas supply bottles~~diesel generator control rooms. Each area has low post LOCA radiation levels. The dose evaluations are within regulatory guidelines.

Access to ~~vital~~ post-accident access areas through out the reactor building/control building/~~turbine~~ electrical building complex is controlled via the service building. Entrance to the service building and access to the other areas are controlled via double locked secured entryways. Access to the reactor building is via two specific routes, one for clean access and the second for controlled access. During an event such as a design basis accident, the main control ~~building~~ room area is maintained under filtered HVAC at positive pressure with respect to the environment. Air infiltration is minimized by positive flow via double entry ways. Therefore, radiation exposure is limited to gamma shine from the reactor building, turbine building, main steam line access corridor, and ~~skyline~~skyshine.

During a design basis accident event, access to ~~nitrogen bottles and~~ monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained

under filtered positive pressure so personal protection equipment (radiation protection suits, breathing gear, etc.) is required in the access corridor. Primary contamination would occur from leakage through primary containment ~~sampling locations~~. This pathway is considered minimal and minor contamination under even the most adverse conditions is expected. Access to the IC pool refill nozzles is from outside ~~and would not likely require special breathing gear, etc~~.

The reactor building ~~vital~~ post-accident access areas are all located off the controlled access way, except for the SLCS pump room; ~~and~~ contamination is limited to air infiltration from the accident environment and penetration leakage from primary containment sampling locations. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from primary containment ~~sampling locations~~. These sources are considered minimal including the stack monitor room, which contains only instrumentation with their associated penetrations for monitoring stack effluent.

The actions beyond 72 hours and actions for long term post-accident recovery are listed in Table 12.3-11.

Figures 12.3-74 through 12.3-90 show the post-accident access areas ands personnel access and egress routes.

### 12.3.6 Post-Accident Radiation Zone Maps and Mission Doses

The airborne radiation source term after a LOCA, as indicated in Chapter 15, is taken as input data using the methodology of Regulatory Guide 1.183. The post-accident dose rates for each time interval at each post-accident access area and room are shown in Tables 12.3-12 and 12.3-13.

The post-accident radiation zone maps for the post-accident access areas in the reactor building, control building, electrical building, and service building are presented in Figures 12.3-43 through 12.3-51i. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. ~~These dose rates do not include the airborne contribution in the reactor building. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.~~

The radiation "mission" dose in performing long-term recovery actions in the post-accident access areas have been calculated and the results are listed in Tables 12.3-14 through 12.3-17. The following criteria are considered:

- The long-term recovery actions in the post-accident access areas are expected to be required well after 72 hours. However, the mission doses have been conservatively calculated as if the mission occurs at 72 hours.
- Both air submersion and inhalation airborne doses are taken into account.
- Round trip travel and occupancy times are taken into account. The mission doses are calculated based on the dose rates of the various rooms.
- A dose reduction factor is assumed by using a pressure demand self-contained breathing apparatus.

Doses for each activity are below the 0.05 Sv (5 rem) GDC 19 limit. Post-accident radiation doses in the Control Room are also described in DCD Tier 2 Section 15.4, and are also below the

0.05 Sv (5 rem) GDC 19 limit. The Technical Support Center (TSC) is designed with the same radiological controls as the Control Room and accordingly the expected post-accident radiation doses are also below the 0.05 Sv criterion.

Although continuous occupancy is not required in the Health Physics Facility (counting room), the cumulative post-accident dose is less than the 0.05 Sv (5 rem) GDC 19 limit assuming continuous occupancy. The average dose in the Count Room is less than 15 mrem/hr which meets the NUREG-0737 II.B.2 requirement for continuous occupancy.

## 12.3.7 COL Information

### *12.3-1-H Facility Design Features (Deleted)*

### *12.3-2-A Operational Considerations*

Airborne radiation monitoring operational considerations such as the procedures for operations and calibration of the monitors, as well as the placement of the portable monitors, are the COL applicant's responsibility (Subsection 12.3.4).

### *12.3-3-A Controlled Access*

Controlled access to "Very High Radiation Areas" is provided by the COL applicant (Subsection 12.3.1.3).

## 12.3.8 References

12.3-1 U.S. Atomic Energy Commission, "Reactor Shielding for Nuclear Engineers," TID-25951, 1973.

12.3-2 U.S. Department of Commerce, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV," NSRDS-NBS20, August 1969.

12.3-3 U.S. Department of Health, Education, and Welfare, "Radiological Health Handbook," Revised Edition, January 1970.

12.3-4 U.S. Atomic Energy Commission, "Reactor Handbook, Volume III, Part B," 1962.

12.3-5 Lederer, Hollander, and Perlman, "Table of Isotopes," Sixth Edition (1968).

12.3-6 General Electric Company, "Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source", APEX-510, November 1958.

12.3-7 U.S. Atomic Energy Commission, "Reactor Physics Constants, Second Edition," ANL-5800, July 1963.

12.3-8 Brookhaven National Laboratory, "ENDF/B-III and ENDF/B-IV Cross Section Libraries".

12.3-9 Oak Ridge National Laboratory, "PDS-31 Cross Section Library".

12.3-10 "DLC-7, ENDF/B Photo Interaction Library".

12.3-11 USNRC, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.

## Table 12.3-11

## Beyond 72 hour and long term post accident recovery actions access requirements

| Room/ | Long term post accident action required |
|---|---|
| 1311: Electrical Equipment Division 1 | Extend systems functionality beyond 72 hr. and long term recovery actions |
| 1313: Remote Shutdown Control Panel Room Division 1 | Short presence for back-up or cross checking actuations |
| 1321: Electrical Equipment Division 2 | Extend systems functionality beyond 72 hr. and long term recovery actions |
| 1323: Remote Shutdown Control Panel Room Division 2 | Short presence for back-up or cross checking actuations |
| 1331: Electrical Equipment Division 3 | Extend systems functionality beyond 72 hr. and long term recovery actions |
| 1341: Electrical Equipment Division 4 | Extend systems functionality beyond 72 hr. and long term recovery actions. |
| 1703: Standby Liquid Control Pump Room | Long term boron refilling |
| 3275: Control Room | Continuous presence for post accident operation and long term recovery actions |
| 3301: Nonsafety-DCIS Room A | Long term recovery of auxiliary functions |
| 3302: Nonsafety-DCIS Room B | Long term recovery of auxiliary functions |
| 5156: Diesel Generator Control Room A | Long term AC power supply recovery actions |
| 5166: Diesel Generator Control Room B | Long term AC power supply recovery actions |
| 5180: Technical Support Center | Continuous presence for emergency support actions |
| Health Physics Facility (Counting Room) | Continuous presence for health physics and sampling measurements support actions |
| Outside: IC/PCC and fuel pool refill valve | Long term refilling/makeup of IC/PCC pools |

# Table 12.3-12

## Dose Rates in the Post-Accident Access Areas

| Room | | Dose rate (mSv/hr) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | 0-0.5 hr | 0.5-2 hr | 2-8 hr | 8-24 hr | 24-72 hr | 3-4 days | 4-7 days | 7-30 days |
| Reactor building | | | | | | | | | |
| Electrical equipment division 1 | 1311 | 0.54 | 11.82 | 20.14 | 12.43 | 10.23 | 9.55 | 6.83 | 0.38 |
| Remote shutdown control panel room division 1 | 1313 | 0.23 | 5.08 | 8.67 | 5.41 | 4.38 | 4.07 | 2.91 | 0.20 |
| Electrical equipment division 2 | 1321 | 0.52 | 11.36 | 19.36 | 11.96 | 9.83 | 9.17 | 6.56 | 0.37 |
| Remote shutdown control panel room division 2 | 1323 | 0.23 | 5.08 | 8.67 | 5.41 | 4.38 | 4.07 | 2.91 | 0.20 |
| Electrical Equipment division 3 | 1331 | 0.55 | 12.01 | 20.46 | 12.63 | 10.39 | 9.70 | 6.94 | 0.39 |
| Electrical equipment division 4 | 1341 | 0.55 | 12.10 | 20.61 | 12.72 | 10.47 | 9.77 | 6.99 | 0.39 |
| Standby liquid control pump room | 1703 | 12.33 | 269.75 | 459.87 | 284.80 | 233.13 | 217.29 | 155.49 | 9.21 |
| Control building | | | | | | | | | |
| DCIS Room A | 3301 | 0.02 | 0.21 | 0.33 | 0.09 | 0.08 | 0.08 | 0.06 | 0.02 |
| DCIS Room B | 3302 | 0.02 | 0.21 | 0.32 | 0.08 | 0.08 | 0.08 | 0.05 | 0.02 |
| Electrical building | | | | | | | | | |
| Diesel generator control Room A | 5156 | 0.01 | 0.11 | 0.18 | 0.05 | 0.04 | 0.04 | 0.03 | 0.01 |
| Diesel generator control Room B | 5166 | 0.01 | 0.11 | 0.18 | 0.05 | 0.04 | 0.04 | 0.03 | 0.01 |
| Counting room | | | | | | | | | |
| Counting room | -- | 0.02 | 0.23 | 0.36 | 0.09 | 0.09 | 0.09 | 0.06 | 0.02 |
| Outside | | | | | | | | | |
| IC/PCCS and fuel refill valve | -- | 0.72 | 8.05 | 12.60 | 3.19 | 2.96 | 3.08 | 2.10 | 0.69 |

# Table 12.3-13

## Dose Rates for the Access Ways to Post-Accident Access Areas

| Room | | Dose rate (mSv/hr) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | 0-0.5 hr | 0.5-2 hr | 2-8 hr | 8-24 hr | 24-72 hr | 3-4 days | 4-7 days | 7-30 days |
| Reactor building | | | | | | | | | |
| Stairwell B | 1191 | 18.22 | 397.80 | 677.96 | 418.34 | 344.40 | 321.44 | 230.04 | 12.73 |
| Controlled equipment removal access room | 1308 | 0.32 | 6.83 | 11.64 | 7.23 | 5.89 | 5.49 | 3.93 | 0.25 |
| Corridor A Division 3 | 1730 | 19.11 | 417.15 | 710.91 | 438.51 | 361.21 | 337.18 | 241.30 | 13.26 |
| Control building | | | | | | | | | |
| Stairwell B | 3192 | 0.02 | 0.16 | 0.25 | 0.07 | 0.06 | 0.06 | 0.04 | 0.01 |
| Corridor | 3200 | 0.02 | 0.16 | 0.25 | 0.07 | 0.06 | 0.06 | 0.04 | 0.01 |
| Airlock vestibule B | 3277 | 0.01 | 0.07 | 0.11 | 0.03 | 0.03 | 0.03 | 0.02 | 0.01 |
| Corridor | 3300 | 0.02 | 0.18 | 0.27 | 0.07 | 0.06 | 0.07 | 0.05 | 0.02 |
| Electrical building | | | | | | | | | |
| Corridor | 5102 | 0.02 | 0.19 | 0.33 | 0.17 | 0.11 | 0.30 | 0.12 | 0.03 |
| Diesel generator A Room | 5154 | 0.03 | 0.32 | 0.51 | 0.13 | 0.12 | 0.12 | 0.08 | 0.03 |
| Diesel generator B Room | 5164 | 0.03 | 0.32 | 0.51 | 0.13 | 0.12 | 0.12 | 0.08 | 0.03 |
| Stairwell C | 5194 | 0.01 | 0.11 | 0.16 | 0.04 | 0.04 | 0.04 | 0.03 | 0.01 |
| Tunnel Access | | | | | | | | | |
| Clean personnel access tunnel | 9101 | 0.03 | 0.35 | 0.54 | 0.14 | 0.13 | 0.13 | 0.09 | 0.03 |
| Controlled personnel Access tunnel | 9201 | 0.02 | 0.17 | 0.26 | 0.07 | 0.07 | 0.07 | 0.05 | 0.02 |
| Service building | | | | | | | | | |
| Break-area 1 | -- | 0.02 | 0.20 | 0.32 | 0.08 | 0.07 | 0.08 | 0.05 | 0.02 |
| Break-area 2 | -- | 0.01 | 0.16 | 0.25 | 0.06 | 0.06 | 0.06 | 0.04 | 0.01 |
| Break-room 1 | -- | 0.02 | 0.17 | 0.26 | 0.07 | 0.06 | 0.06 | 0.04 | 0.01 |
| Break-room 2 | -- | 0.02 | 0.18 | 0.28 | 0.07 | 0.07 | 0.07 | 0.05 | 0.02 |
| Corridor 1 | -- | 0.02 | 0.25 | 0.39 | 0.10 | 0.09 | 0.10 | 0.06 | 0.02 |
| Corridor 2 | -- | 0.02 | 0.23 | 0.36 | 0.09 | 0.09 | 0.09 | 0.06 | 0.02 |
| Corridor 3 | -- | 0.02 | 0.21 | 0.32 | 0.08 | 0.08 | 0.08 | 0.05 | 0.02 |
| Lobby | -- | 0.01 | 0.14 | 0.22 | 0.06 | 0.05 | 0.05 | 0.04 | 0.01 |
| Stairs 1 | -- | 0.01 | 0.14 | 0.21 | 0.05 | 0.05 | 0.05 | 0.03 | 0.01 |
| Stairs 2 | -- | 0.01 | 0.14 | 0.21 | 0.05 | 0.05 | 0.05 | 0.03 | 0.01 |

## Table 12.3-14

## Reactor Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| **Mission to room 1311: Electrical equipment division 1** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 17.15 | 0.392 | 4.77E-04 |
| Electrical equipment division 1 | 1311 | 69.99 | 1.276 + 5 (1) | 1.07E+00 |
| Clean personal access tunnel | 9101 | 17.15 | 0.365 | 4.45E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Stairwell B | 3192 | 15.03 | 0.593 | 5.98E-04 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.08E+00** |
| **Mission to room 1321: Electrical equipment division 2** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3200 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 14.85 | 0.350 | 4.26E-04 |
| Electrical equipment division 2 | 1331 | 30.325 | 0.553 | 9.57E-02 |
| Controlled equipment removal access room | 1308 | 8.376 | 0.153 | 1.50E-02 |
| Electrical equipment division 2 | 1321 | 56.512 | 1.030 + 5 (1) | 9.88E-01 |
| Controlled equipment removal access room | 1308 | 8.376 | 0.153 | 1.50E-02 |
| Electrical equipment division 2 | 1331 | 30.325 | 0.553 | 9.57E-02 |
| Clean personal access tunnel | 9101 | 14.85 | 0.323 | 3.94E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Stairwell B | 3192 | 15.03 | 0.593 | 5.98E-04 |

# Table 12.3-14

## Reactor Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.22E+00** |
| **Mission to room 1331: Electrical Equipment Division 3** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 14.85 | 0.350 | 4.26E-04 |
| Electrical equipment division 3 | 1331 | 60.65 | 1.105 + 5 (1) | 1.06E+00 |
| Clean personal access tunnel | 9101 | 14.85 | 0.323 | 3.94E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Stairwell B | 3192 | 15.03 | 0.593 | 5.98E-04 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.06E+00** |
| **Mission to room 1341: Electrical equipment division 4** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 17.15 | 0.392 | 4.77E-04 |
| Electrical Equipment Division 1 | 1311 | 35.00 | 0.638 | 1.09E-01 |
| Electrical equipment division 4 | 1341 | 70.94 | 1.293 + 5 (1) | 1.09E+00 |
| Electrical Equipment Division 1 | 1311 | 34.995 | 0.638 | 1.09E-01 |
| Clean personal access tunnel | 9101 | 17.15 | 0.365 | 4.45E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |

# Table 12.3-14

## Reactor Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| Stairwell B | 3192 | 15.03 | 0.593 | 5.98E-04 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.31E+00** |
| **Mission to room 1313: Remote shutdown control panel room division 1** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 17.15 | 0.392 | 4.77E-04 |
| Electrical equipment division 1 | 1311 | 26.804 | 0.489 | 8.33E-02 |
| Remote shutdown control panel room division 1 | 1313 | 9.924 | 0.181+0.5+5 (1) | 4.14E-01 |
| Electrical equipment division 1 | 1311 | 26.804 | 0.489 | 8.33E-02 |
| Clean personal access tunnel | 9101 | 17.15 | 0.365 | 4.45E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Stairwell B | 3192 | 15.03 | 0.593 | 5.98E-04 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 min time of operation and 0.5 min for opening door are assumed | | **Total dose:** | | **5.88E-01** |
| **Mission to room 1323: Remote shutdown control panel room division 2** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Stairwell B | 3192 | 15.03 | 0.487 | 4.90E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Clean personal access tunnel | 9101 | 14.85 | 0.350 | 4.26E-04 |
| Electrical equipment division 3 | 1331 | 30.325 | 0.553 | 9.57E-02 |
| Controlled equipment removal access room | 1308 | 8.376 | 0.153 | 1.50E-02 |

# Table 12.3-14

## Reactor Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| Electrical equipment division 2 | 1321 | 6.502 | 0.119 | 1.94E-02 |
| Remote shutdown control panel room division 2 | 1323 | 9.384 | 0.671+0.5+5 (1) | 4.14E-01 |
| Electrical equipment division 2 | 1321 | 6.502 | 0.119 | 1.94E-02 |
| Controlled equipment removal access room | 1308 | 8.376 | 0.153 | 1.50E-02 |
| Electrical equipment division 3 | 1331 | 30.325 | 0.553 | 9.57E-02 |
| Clean personal access tunnel | 9101 | 14.85 | 0.323 | 3.94E-04 |
| Corridor | 3200 | 29.25 | 0.533 | 6.75E-04 |
| Stairwell B | 3192 | 15.03 | 0.541 | 5.45E-04 |
| Corridor | 3300 | 6.34 | 0.116 | 1.25E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 min time of operation and 0.5 min for opening door are assumed | | **Total dose:** | | **6.81E-01** |
| **Mission to room 1703: Standby liquid control pump room** | | | | |
| Lobby | -- | 6.453 | 0.118 | 1.01E-04 |
| Break-area 1 | -- | 23.726 | 0.432 | 5.35E-04 |
| Stairs 1 | -- | 16.342 | 0.502 | 4.13E-04 |
| Break-room 1 | -- | 12.252 | 0.223 | 2.30E-04 |
| Corridor 2 | -- | 49.986 | 0.911 | 1.30E-03 |
| Controlled personnel access tunnel | 9201 | 16.48 | 0.341 | 3.95E-04 |
| Stairwell B | 1191 | 40.07 | 1.544 | 8.86E+00 |
| Corridor A division 3 | 1730 | 11.693 | 0.213 | 1.28E+00 |
| Standby liquid control pump room | 1703 | 11.7 | 0.713+0.5+5 (1) | 2.22E+01 |
| Corridor A division 3 | 1703 | 11.693 | 0.213 | 1.28E+00 |
| Stairwell B | 1191 | 40.07 | 1.056 | 6.06E+00 |
| Controlled personnel access tunnel | 9201 | 16.48 | 0.273 | 3.17E-04 |
| Corridor 3 | -- | 38.301 | 0.698 | 8.81E-04 |
| Break-room 2 | -- | 10.048 | 0.183 | 2.00E-04 |
| Stairs 2 | -- | 16.415 | 0.605 | 4.99E-04 |
| Break-area 2 | -- | 5.189 | 0.095 | 9.39E-05 |
| Lobby | -- | 6.453 | 0.118 | 1.01E-04 |
| (1) 5 min time of operation and 0.5 min for opening door are assumed | | **Total dose:** | | **3.97E+01** |

## Table 12.3-15

## Control Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| **Mission to room 3301: DCIS Room A** | | | | |
| Lobby | -- | 6.456 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 42.058 | 0.767 | 8.26E-04 |
| DCIS Room A | 3301 | 55.722 | 1.097 + 5 (1) | 8.04E-03 |
| Corridor | 3300 | 42.058 | 0.767 | 8.26E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.456 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.34E-02** |
| **Mission to room 3302: DCIS Room B** | | | | |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Corridor | 3300 | 40.073 | 0.730 | 7.87E-04 |
| DCIS Room B | 3302 | 65.516 | 1.276 +5 (1) | 7.96E-03 |
| Corridor | 3300 | 40.073 | 0730 | 7.87E-04 |
| Corridor 1 | -- | 63.23 | 1.152 | 1.76E-03 |
| Lobby | -- | 6.45 | 0.118 | 1.01E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **1.33E-02** |

## Table 12.3-16

## Electrical Building Post-Accident Access Area - Radiation Mission Doses at 72 hr

| Path | Room | Walked distance (m) | Time (min) | Dose (mSv) |
|---|---|---|---|---|
| **Mission to room 5156: Diesel generator room A** | | | | |
| Diesel Generator Room A | 5154 | 9.002 | 0.164 | 3.25E-04 |
| Diesel Generator A Room | 5156 | 21.382 | 0.390 + 5 (1) | 3.75E-03 |
| Diesel Generator Room A | 5154 | 9.002 | 0.164 | 3.25E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **4.40E-03** |
| **Mission to room 5166: Diesel generator room B** | | | | |
| Diesel Generator Room A | 5164 | 9.051 | 0.165 | 3.26E-04 |
| Diesel Generator B Room | 5166 | 21.38 | 0.390 + 5 (1) | 3.75E-03 |
| Diesel Generator Room A | 5164 | 9.051 | 0.165 | 3.26E-04 |
| (1) 5 minutes time of operation is assumed | | **Total dose:** | | **4.40E-03** |

## Table 12.3-17

## Outside Yard Area - Post-Accident Radiation Mission Dose at 72 hr

| Path | Walked distance (m) | Time (min) | Dose (mSv) | |
|---|---|---|---|---|
| IC/PCC and fuel refill valve | | | | |
| IC/PCC and fuel refill valve | 0 | 20 | 2.47E-01 | 72 h |

**Table 1A-1**

**TMI Action Plan Items**

| Regulation | TMI Item | Description | ESBWR Resolution | Associated Location(s) |
|---|---|---|---|---|
| | | | Equipment and Floor Drainage System (EFDS) with two normally closed valves during reactor power operation. These vent and purge lines are not required to assure natural circulation core cooling. | |
| 10 CFR 50. 34(f)(2)(vii) | II.B.2 | Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect equipment from the radiation environment. | The Alternate Source Term (AST) contained in Reg. Guide 1.183 has superseded the TID-14844 source term. The AST is used for radiation design issues in the ESBWR.<br>Reviews of ESBWR spaces requiring post-accident access reveals that each area has low post LOCA radiation levels.<br>A review of the radiation and shielding of the ESBWR post-accident operations has been made. It has been found that there is adequate access to vital areas and that safety equipment is adequately protected.<br>An evaluation of post-accident radioactive sources concluded that the ESBWR design limits potential radiation exposure from accidents both to plant personnel and to the public by the use of passive safety features and holdup in the containment.<br>Potential releases in the radwaste building are contained by isolating the radwaste building atmosphere and containing any water releases in the building, which is | Appendix 1B, 3.1.2, 3.11, 12.3.5, 12.3.6, Figures 12.3-43 through 12.3-51, and 15.4.1.3. |

**Table 1A-1**

**TMI Action Plan Items**

| Regulation | TMI Item | Description | ESBWR Resolution | Associated Location(s) |
|---|---|---|---|---|
| | | | seismically qualified and designed to prevent any potential water releases from high activity areas. Additional details relating to plant radiation sources can be found in Section 12.2.<br>The locations requiring access to mitigate the consequences of an accident during the post-accident period are the control room, the technical support center, electrical equipment rooms for divisions 1, 2, 3 and 4, the Standby Liquid Control system pump room, nonsafety-related Distributed Control and Information System (DCIS) rooms, the remote shutdown panel rooms, ~~the primary containment sample station (process sample system),~~ the health physics facility (counting room), diesel generator control rooms, and the IC/PCC and fuel pool refill valves~~, and the nitrogen gas supply bottles~~. Each area has low post-LOCA radiation levels. The results of the evaluations are reflected in the radiation zone maps (Figures 12.3-43 through 12.3-51) and demonstrate that personnel doses will be within regulatory guidelines.<br>Areas in ~~T~~the reactor building requiring post-accident access ~~vital areas~~ are all | |

**Table 1A-1**

**TMI Action Plan Items**

| Regulation | TMI Item | Description | ESBWR Resolution | Associated Location(s) |
|---|---|---|---|---|
| | | | located off the controlled access way and contamination is limited to air infiltration from the containment and contaminated systems. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from monitoring systems such as the Process Sampling System (PSS).<br>An environmental qualification program for safety-related mechanical and electrical equipment to demonstrate their capability to perform their required functions when exposed to the environmental conditions (including accident and post-accident conditions) in their respective locations is described in Section 3.11. Radiation shielding is designed to keep radiation doses to equipment below levels at which disabling radiation damage occurs | |
| 10 CFR 50. 34(f)(2)(viii) | II.B.3 | Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the | The ESBWR Containment Monitoring System (CMS) and Process Sampling System (PSS) provide the required capability to obtain and analyze highly radioactive post accident samples from the reactor coolant system, the containment sump, and the containment atmosphere.<br>The Process Sampling System described in | 7.5.2, 7.5.3, 9.3.2, and 11.5. |

(2) Based upon the accident source terms of Regulatory Guides 1.183 and 1.7 and Standard Review Plan 15.0.1 (References 1B-2, 1B-3 and 1B-4, respectively) including normal operations, exposures for equipment requiring post-accident access are enveloped based upon the Table below:

| Area | Gamma (Gy) | Beta (Gy) |
|---|---|---|
| Containment | $2x10^6$ | $2x10^7$ |

Each actual area is environmentally qualified to the area specific envelope as defined in Appendix 3H.

(3) It is not necessary for operating personnel to have access to any place other than the control room, technical support center, electrical equipment rooms for divisions 1, 2, 3 and 4, the Standby Liquid Control (SLC) system pump room, nonsafety-related Distributed Control and Information System (DCIS) rooms, remote shutdown panels, ~~primary containment sampling locations,~~ health physics facility (counting room), diesel generator control rooms, ~~High Pressure Nitrogen Supply System (HPNSS) bottles,~~ and ~~external connections for~~ Isolation Condenser/Passive Containment Cooling (IC/PCC) and fuel pools refill valves ~~makeup~~ as described in Appendix 19A (19A.1, Criterion B) to operate the equipment of interest during the 100-day environmental qualification period required by Appendix 3H. The control room~~,~~ and the technical support center ~~and sample analysis areas~~ are designed to be physically accessible post-accident. The ability to occupy the technical support center ~~and sample analysis areas~~ post-accident is subject to the presence of acceptable radiation levels at those locations.

(4) Access to radwaste control panels or equipment is not required, but the Radwaste Building (RW) is accessible, since containment sump discharges are isolated. Thus, fission products are not transported to the Radwaste Building. The ESBWR does not have a containment isolation reset control 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 to corridors in the upper Reactor Building (RB) (see post-accident radiation zone maps in Subsection 12.3.6). This is based on radiation shine from the containment. While not necessary to maintain safe shutdown, such access is useful in extending system functionality and facilitating plant recovery.

(6) The safety-related power supplies identified in Table 1B-5 are accessible. However, access is not necessary. Nonsafety-related diesel generators are also available and accessible to provide power.

## 1B.3 CONTAINMENT DESCRIPTION AND POST-ACCIDENT OPERATIONS

### 1B.3.1 Description of Containment

The ESBWR design includes many features to assure that personnel occupancy is not unduly limited and that safety-related equipment is not degraded by post-accident radiation fields.

These features are detailed in other DCD locations. Consequently, only a brief summary description and a reference to other DCD locations are provided here for emphasis.

The configuration of the pressure suppression containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the containment atmosphere following an accident is thereby substantially reduced compared to the Reg. Guide 1.183 source terms. The Passive Containment Cooling System (PCCS) condensing function contributes to reduce many of the airborne fission products.

Containment leakage is limited to less than 0.4% ~~one half percent~~ of the weight in the containment free volume per day.

### 1B.3.2 Post-Accident Access of Areas and Systems

This section addresses any area that may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas that must be accessible after an accident are the control room and technical support center. ~~The nonsafety-related High Pressure Nitrogen Supply System (HPNSS) nitrogen supply bottles are available for use to operate containment isolation valves inside containment if necessary in support of long-term post-accident actions.~~

Areas requiring post-accident access also include consideration (in accordance with NUREG-0737, II.B.2) of the containment isolation reset control area, manual ECCS alignment area, motor control centers and radwaste control panels. However, the ESBWR 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 or are not applicable for the passive ECC systems. Areas requiring post-accident access that are normally areas of mild environment allowing unlimited access are not reviewed for access.

Systems specific to the ESBWR that may require post-accident access are those for long-term core cooling, fission product control and combustible gas monitoring, as well as the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

### 1B.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 10 CFR 50.34(a) guidelines.

Safety-related systems are required for scram and to achieve a safe shutdown condition. However, they are not necessarily needed to maintain safe shutdown. The systems identified in Section 1B.5 are the systems 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 ~~manual nitrogen reserve supply valves and~~ manual external connections for the IC/PCC and fuel pools makeup.

## Table 1B-3

## Post-Accident Containment Systems and Auxiliaries

| Equipment | MPL | Location |
|---|---|---|
| ~~**High Pressure Nitrogen Supply System (HPNSS)**~~ | | |
| ~~Nitrogen Storage Bottles~~ | ~~P54~~ | ~~By Valve Room (RB)~~ |
| ~~Supply Pressure~~ | ~~P54~~ | ~~By Valve Room (RB)~~ |
| **Containment Monitoring System (CMS)** | | |
| Hydrogen, Oxygen Elements | T62 | CMS Rooms (RB) |
| Gas Measurement | T62 | CMS Rooms (RB) |
| Gas Elements | T62 | CMS Rooms (RB) |
| Drywell Gas Valve | T62 | CMS Rooms (RB) |
| Wetwell Gas Valve | T62 | CMS Rooms (RB) |
| Gas Supply | T62 | CMS Rooms (RB) |
| **Passive Containment Cooling System (PCCS)** | | |
| Condenser Units | T15 | IC/PCC Pools (RB) |
| Condensate Discharge | T15 | GDCS Pools (C) |
| Non-condensable Gas Discharge | T15 | Suppression Pool Wetwell (C) |

MPL — Master Parts List Number designated for the system

(C) — Containment

(RB) — Reactor Building