GULF STATES UTILITIES COMPANY

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

River Bend Station Units 1 & 2
Docket Nos. 50-458/50-459

Enclosed for your review are Gulf States Utilities Company's responses to Draft Safety Evaluation Report (DSER) open items identified by the Nuclear Regulatory Commission's Radiological Assessment Branch (RAB). In addition, the Licensing Review Group-II (LRG) positions 1, 2, 3, and 4-RAB are endorsed herein. Attachment 1 is a summary listing of the items discussed in Attachment 2. Attachment 2 provides the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerely,

J. E. Booker

J. E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/RJK/kt

Attachments-40 copies

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Attachment 1

RAB Open DSER Items

Item	DSER Section	Related Question #	Subject	FSAR Change
1	12.1.2 pg.12-3 12.3.1 pg.12-5	Q#471.13	Design and Construction ALARA Review	Enclosure 1
2	12.2.2 pg.12-5	Q#471.20	Accident Source Terms-II.B.2	Amendment 7 & Enclosure 2
3	12.2.2 pg.12-5	Q#471.11	Tabulation of Normal Expected Airborne Radioactivity Concentrations.	Enclosure 3
4	12.3.1 pg.12-6	Q#471.17	Minimize Corrosion Buildup	Amendment 7
5	12.3.4.2 pg.12-10	Q#471.16	Monitors to Detect 10 MPC-hours	Enclosure 4
6	12.3.4.2 pg.12-10	Q#471.20	High-Range Radioactivity Monitors-II.F.1(3)	Amendment 7 & Enclosure 2
7	12.4 pg.12-10	Q#471.10	Personnel Dose Assessment and Improvements	Amendment 3 & Enclosure 5
8	12.5.1 pg.12-11	Q#471.21	Qualifications of the Radiation Protection/ Chemistry Supervisor and Backup	Amendment 5 & Enclosure 6
9	12.5.1 pg.12-11	Q#471.22	Radiation Protection/Chemistry Supervisor Direct Access to the Project Manager	Amendment 5
10	12.5.2 pg.12-13	Q#471.20	Iodine Monitoring-III.D.3.3	Amendment 7 & Enclosure 2
11		Q#471.10	LRG-II 1-RAB Exposure From SRV Actuation	Enclosure 7
		Q#471.11	LRG-II 2-RAB Routine Exposures Inside Containment	Enclosure 7
		Q#471.11	LRG-II 3-RAB Radioactivity During Dryer/Separator Transfers	Enclosure 7
		Q#471.11	LRG-II 4-RAB Shielding of Spent Fuel Transfer Tube and Canal During Refueling	Enclosure 7

Attachment 2
Responses to RAB DSER Open Items

1. DSER, page 12-3-Design and construction ALARA reviews

   Response:

   The RBS design and construction ALARA reviews are discussed in detail in Enclosure 1. The information provided will be incorporated into the FSAR in a future amendment as the response to Question 471.13.

2. DSER, page 12-5-Calculated accident source terms; TMI Action Plan Item II.B.2

   Response:

   A response was provided in Amendment 7 to Question 471.20 in Appendix 1A. A revised response is provided in Enclosure 2.

3. DSER, page 12-5-Tabulation of normal expected airborne radioactivity concentration

   Response:

   A revision to FSAR Section 12.2.2, "Airborne Radioactive Material Sources" is provided in Enclosure 3. A description and tabulation of normal expected inplant airborne radioactivity concentrations,as a result of equipment leakage in pump rooms and operating areas,will be incorporated into the FSAR in a future amendment as the response to Question 471.11.

4. DSER, page 12-5-Minimize corrosion buildup

   Response:

   This item is addressed in the response to FSAR Question 471.17, Chapter 12, which was submitted to the NRC in FSAR Amendment 2.

5. DSER, page 12-10-Monitors to detect 10mpc-hours of particulate or iodine radioactivity

   Response:

   A description is provided in revised Section 12.3.4.2.5 (Enclosure 4) on area radiation monitoring for detecting 10mpc-hours of airborne particulate radioactivity. The description and referenced Tables will be incorporated into the FSAR in a future amendment as the response to Question 471.16.

RAB

Attachment 2 (Cont'd)

6. DSER, page 12-10-Containment high-range radiation monitors; TMI Action Plan Item II.F.1-3

Response:

A response was provided in Amendment 7 to Question 471.20 in Appendix 1A (Enclosure 2). The response referenced Section 7.5 of the FSAR. The containment high-range radiation monitor is listed in Tables 7.5-1, page 2 of 12, and 7.5-2, page 5 of 13.

7. DSER, page 12-10-Personnel dose assessment and improvements

Response:

A response to Question 471.10 was provided in Amendment 3 which referenced Sections 12.1.2.5 and 12.4.1 (Enclosure 5). The occupational dose assessment for RBS will be in accordance with Regulatory Guide 8.19. The results of the dose assessment and a listing of any plant improvements will be provided by March 31, 1984. The dose assessment will also include exposure from SRV actuation consistent with LRG-II position 1-RAB.

8. DSER, page 12-11-Qualifications of the RBS Radiation Protection/Chemistry Supervisor and his backup

Response:

The qualifications of the RBS Radiation Protection/Chemistry Supervisor are provided in Section 13.1.3.2. The Radiation Protection Supervisor is the normal backup; however, at this time, the Radiation Protection Supervisor's position has not been filled. The qualification for the normal backup will possess the required qualifications as specified in GSU's response to Question 471.21 (Enclosure 6). This response will be incorporated into the FSAR in a future amendment as part of the response to Question 471.21.

9. DSER, page 12-11-Radiation Protection/Chemistry Supervisor direct access to the Project Manager

Response:

This item is addressed in the response to FSAR Question 471.22, Chapter 13, which was submitted to the NRC in FSAR Amendment 5.

10. DSER, page 12-13-Iodine monitoring, TMI Action Plan Item III.D.3.3

Response:

This item is addressed in GSU's response to FSAR Question 471.20, Appendix 1A, which was submitted to the NRC in FSAR Amendment 7. Additional information is provided in Enclosure 2.

11. The endorsement to LRG-II positions 1-RAB through 4-RAB is provided below. The discussion of these items are provided in Enclosures 5 and 7.

Item	Title	Endorsed	FSAR Discussion
1-RAB	Exposure From SRV Actuation	Yes	12.4.1, Q471.10
2-RAB	Routine Exposures Inside Containment	Yes	12.3.2.2.2, Q471.11
3-RAB	Radioactivity During Dryer/Separator Transfers	Yes	12.5.3.2.1, Q471.11
4-RAB	Shielding of Spent Fuel Transfer Tube and Canal During Refueling	Yes	12.3.2.2.2, Q471.11

QUESTION 471.12 (12.1)

As specified in Regulatory Guide 1.70, Section 12.1.2, with regard to the radiation protection reviews to assure that ALARA objectives are being met in the design process and during construction (including field run piping):

a. Identify by title the individual(s) responsible for the radiation protection design review, and describe how they relate to the individual responsible for the overall design.

b. Provide a breakdown by title of radiation protection personnel who have been or will be participating in these reviews, tabulating the health physics education and experience required of each.

c. Describe formal arrangements and procedures for assuring that adequate independent radiation protection reviews are performed throughout the design and construction processes, and that adequate records are kept to document the completion of each such review.

RESPONSE

The response to this request $\boxed{\text{will be provided by June 1982.}}$ is provided in revised Section 12.1.1 and 12.1.2.

CHAPTER 12

RADIATION PROTECTION

This chapter provides information relevant to the radiation protection features of the facility and equipment design, plans and practices to accomplish radiological control, and estimates of occupational radiation exposure to operating and contractual personnel during normal operation and anticipated operational occurrences.

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

12.1.1 Policy Considerations

It is the policy of GSU to keep all radiation exposure of personnel within the limits established by the federal government as set forth in Title 10 of the Code of Federal Regulations. Administrative procedures and practices maintaining radiation exposure of personnel as low as is reasonably achievable (ALARA) are delineated herein. The specific River Bend Station positions on Division 8 Regulatory Guides, including Regulatory Guides 8.8 and 8.10, 1
are provided in Section 1.8. The bases for these positions are contained in the discussions found in this section and throughout Chapter 12. ← Insert A

The purpose of the ALARA program is to maintain the radiation exposure of plant personnel as far below the regulatory limits as is reasonably achievable.

The Vice President - Nuclear Operations and Technical Systems reports to the Senior Vice-President - River Bend Nuclear Group (Section 13.1) and is responsible for implementing the ALARA program by: 1) ensuring that the resources needed to achieve the goals and objectives of the ALARA program are available, and 2) ensuring that the authority for developing the procedures and practices by which the goals and objectives are achieved, measured, and evaluated is delegated to the appropriate personnel (Section 13.1).

The River Bend Station Plant Manager reports to the Vice President - Nuclear Operations and Technical Systems and is responsible for the development of plans for review and approval of the radiation protection programs to ensure that radiation exposure of personnel is maintained ALARA.

The River Bend Station Plant Manager also assumes responsibility of the overall effectiveness of the ALARA.

INSERT A

The GSU ALARA policy is consistent with the guidelines in Section C.1 of Regulatory Guide 8.8 and Regulatory Guide 8.10 in establishing, organizing and operating an effective ALARA program.

Chemistry

program at the time of fuel loading. This responsibility is carried out through the Radiation Protection Supervisor.

The Radiation Protection Supervisor is responsible for the development of required radiation protection programs including the ALARA program. He provides technical assistance in conducting the ALARA program and reports to management on the results of the ALARA program.

The Radiation Engineer reports to the Radiation Protection Supervisor and is responsible for coordinating the established radiation protection program. He is responsible for maintaining radiation exposure of personnel ALARA.

Insert B

Further, it is the responsibility of each supervisor to enforce the requirements for keeping radiation exposure ALARA, and the responsibility of each individual to comply with these requirements.

12.1.2 Design Considerations

This section discusses the general methods and features that implement the policy considerations of Section 12.1.1. Detailed provisions for maintaining personnel exposures ALARA are presented in Sections 12.3.1, 12.3.2, and 12.5.3.

Insert C

12.1.2.1 General Design Considerations for ALARA Exposures

The general design considerations and methods employed to maintain in-plant radiation exposures ALARA have two objectives:

1. Minimizing the amount of time plant personnel spend in radiation areas

2. Minimizing radiation levels in routinely occupied plant areas and in the vicinity of plant equipment expected to require the attention of plant personnel.

Both equipment design and arrangement are considered in maintaining exposures ALARA during plant operations, including: normal operation, radwaste handling, normal maintenance, corrective maintenance, refueling, inservice inspection, and other events of moderate frequency and certain infrequently occurring events.

INSERT B

The Radiation Protection/Chemistry Supervisor is responsible for the development of the required radiation protection programs including the ALARA program. He has direct access to the River Bend Station Plant Manager in all radiation protection matters. He reports to GSU management on the effectiveness of the ALARA program. The Radiation Protection Supervisor reports to the Radiation Protection/Chemistry Supervisor. The Radiation Protection Supervisor is also responsible for the development of the ALARA program. He provides technical assistance in conducting the ALARA program and in improving the effectiveness of the ALARA program.

The Radiation Engineer reports to the Radiation Protection Supervisor and is responsible for coordinating and implementing the established radiation protection ALARA program. He is responsible for maintaining occupational radiation exposure of personnel ALARA. He is responsible for conducting design reviews for plant modifications, plant procedures and work practices to assure that ALARA objectives are being met.

INSERT C

The ALARA philosophy was used during the design and construction stages of the River Bend Station by General Electric and Stone & Webster engineers in their continuous reviews of plant design and equipment selection for River Bend Station. Operating experience at BWR's has been incorporated into the River Bend Station design for equipment selection and plant layout to help maintain occupational radiation exposure as low as reasonably achievable.

During the design and construction stages, GSU has performed an independent ALARA design review of River Bend Station. The GSU design review of plant systems and facilities was organized and developed within the Radiation Protection Section under the direction of the Radiation Protection Supervisor. An ALARA Coordinator was assigned the responsibility to develop, perform and document the ALARA design reviews. The ALARA Coordinator directed the actual design reviews through the ALARA Review Group which consisted of GSU Radiation Protection personnel, such as, the Radiation Protection Foreman and Radiation Engineer, and outside consultants experienced in radiation protection, plant design, operation and maintenance. The ALARA Review Group performed and documented their design reviews in a report to the ALARA Committee. The ALARA Committee is chaired by the Radiation Protection Supervisor and consists of GSU representatives from the Operations, Maintenance, Chemistry, Radiation Protection, Technical Staff and Nuclear Plant Engineering Sections. Stone & Webster and General Electric engineers were consulted by the ALARA Committee for design information and for implementing design changes. The ALARA Committee was responsible for reviewing and acting on all the ALARA concerns and recommendations of the ALARA Review Group based on their reports of the plant systems and facility reviews. The actions of the ALARA Committee are approved by the River Bend Station Plant Manager for all design modifications and administrative controls recommended by the Committee to maintain occupational radiation exposure ALARA. The ALARA Committee actions are documented with ALARA Committee Action Items which provide a history for every ALARA concern identified by the ALARA Review Group.

4. Separation of more radioactive equipment from less radioactive equipment

5. Features to minimize crud buildup

6. Coatings applied to surfaces likely to become contaminated to facilitate cleanup.

12.1.2.5 Illustrative Examples of ALARA Improvements

Design improvements, which indicate man-rem reductions during operation and maintenance, have been implemented following ALARA reviews. Some examples follow.

Improvement Based on Dose Assessment

3 GSU is performing and documenting an ALARA design review of River Bend Station. Exposure reducing design features discussed in this chapter, and in Regulatory Guides 8.8 and 8.19 are considered in the design review. An occupational dose assessment is being conducted as a part of this review.

A copy of the dose assessment and a listing of any resulting plant improvements will be provided in a future amendment.

Improvement Based on Operational Experience

Operational experience with radwaste filters, sludge processing, and radwaste solidification systems was utilized in the design and selection of components for systems. Use was also made of studies of operating plant experiences with similar systems to determine system failure rates, down times, number of personnel and man-hours required to repair the systems, and the man-rem associated with repairs.

Operating experience from other units and the ALARA design features of River Bend Station are utilized in the development and revision of station operating and maintenance procedures and instructions to ensure that occupational radiation exposure is maintained ALARA.

Improvement Based on ALARA Design Review

3 GSU is performing and documenting an ALARA design review of River Bend Station. Exposure reducing design features discussed in this chapter, and in Regulatory Guides 8.8 and 8.19 are considered in the design review. An occupational dose assessment is being conducted as a part of this review. Insert D

Insert D (cont'd.)

A copy of the dose assessment and a listing of any resulting plant improvements will be provided in a future amendment. 3

12.1.3 Operational Considerations

In accordance with GSU company policy, the radiation exposure of plant personnel is maintained ALARA by means of the radiation protection program presented in Section 12.5.

The radiation protection training provided at River Bend Station is a primary means of instituting the operational ALARA program. The extent of this training provided each person is at least commensurate with his job

INSERT D

GSU has performed and documented an ALARA design review of River Bend Station during the design and construction stages as described in FSAR Section 12.1.2. During the review the River Bend Station plant facilities and systems were studied in detail to document the design features and any potential design improvements that could minimize the occupational radiation exposure to personnel. The improvements made as a result of the ALARA design review can be categorized as either administrative controls or design changes.

The administrative controls that were implemented to minimize the occupational radiation exposure at River Bend Station included the following:

1. The identification of potential crud traps to the Radiation Protection Section for incorporation into their radiation surveys prior to maintenance or operation activities in areas near these potential hot spots.

2. The identification of the need to emphasize training of maintenance personnel, including hands-on experience in working in the control rod drive (CRD) maintenance area, prior to handling of radioactive CRD's. This is essential due to the high radiation levels associated with CRD maintenance.

3. The identification of training needs for Radiation Protection technicians in the areas of contamination control and radiological hazards with the CRD changeout, transportation and rebuild activities.

4. The use of condensate water throughout the plant for a source of flushing and decontaminating water to minimize the amount of radwaste water generated.

5. The identification of training needs for Operations and Maintenance personnel on the component identification system at River Bend Station to minimize necessary radiation exposure due to identification errors.

6. Operating procedures for the Reactor Water Cleanup System to minimize dumping of reactor coolant water to the condenser to control buildup of corrosion products in condenser and condensate system.

7. The identification of the plant model to Training Department as a useful ALARA tool for pre-job planning.

8. The use of a dedicated set of operators for radwaste operations will improve the efficiency of radwaste processing and thus minimize the maintenance required.

9. Incorporation of adequate flushing in operating procedures for reactor coolant, radwaste liquid and resin transfers to minimize radioactive corrosion product buildup.

10. The identification of training needs for Maintenance personnel on the maintenance activities associated with major components in drywell such as the main steam isolation valves and recirculation pumps in order to minimize radiation exposure in potentially high radiation areas.

11. The control of personnel access to areas near the fuel transfer tube during fuel transfer operations using locked gates, interlocks and radiation monitors.

The design changes that were implemented to minimize the occupational radiation exposure at River Bend Station included the following:

1. Installation of flush connections on the CRD scram discharge volume headers to minimize radiation fields due to corrosion product buildup.

2. Modification of condenser shielding to minimize radiation fields in high traffic areas near the reactor feedwater pumps and instrument racks located in this area.

3. Installation of floor curbs throughout the plant to control the spread of liquid contamination due to leakage from pumps, valves, etc.

4. Installation of flush connections on the liquid radwaste inlet headers to minimize corrosion product buildup.

5. Improved accessibility to plant system components for maintenance and operation activities via platforms, vertical ladders, catwalks, etc.

6. Painting of walls and floors in potentially contaminated areas for ease of decontamination.

7. Use of condensate water for routine sources of flush and decontamination water throughout the plant.

8. Installation of junction boxes for using portable continuous air monitors connected to the Digital Radiation Monitoring System.

9. Design improvements to the Reactor Water Cleanup pump seal design to improve pump reliability.

10. Modification of shielding around reactor water cleanup backwash tank to minimize radiation exposure in adjacent pump cubicle.

11. Relocation of solenoid operators, instruments, condensate and service air isolation valves outside high radiation areas.

12. Installation of remote monitoring equipment for the condensate demineralizer regeneration process equipment.

Question 471.20

Provide the information requested in II.B.2, II.F.1(3) and III.D.3.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements".

RESPONSE

The response to this request is provided in Appendix 1A. In addition, revised Section 12.3.4.1, Tables 12.3-1, 2, and 3 and Table 7.5-1 address the high-range area radiation monitors installed in containment and drywell in response to item II.F.1(3). Revised Section 12.5.2.2.4 describes the portable iodine monitors required in response to item III.D.3.3.

2. Inform the main control room operator of the occurrence and the approximate location of an abnormal radiation increase in nonradiation or low-radiation areas.

3. Comply with the requirements of 10CFR50, Appendix A, GDC 63, for monitoring fuel and waste storage and handling areas.

4. Monitor the new fuel storage area for criticality in compliance with requirements of 10CFR70.24.

5. Comply with the requirements of 10CFR50, Appendix A, GDC 64, to monitor post-accident radiation levels in the reactor containment and drywell.

6. In general, assist in maintaining personnel exposures ALARA.

The area radiation monitoring system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment.

12.3.4.1.1 Area Radiation Monitoring System Design Criteria

The following design criteria are applicable to the area radiation monitoring system:

1. Range - To cover the various ranges anticipated in the plant four different models of the basic instrument with the following scales are provided: 3

   10^{-2} to 10^{4} mrem/hr
   10^{0} to 10^{6} mrem/hr 3
   10^{-1} to 10^{4} R/hr (post-accident vital area monitors)
   10^{0} to 10^{7} rem/hr (for containment and drywell post-accident monitoring).

2. Alarms - Each area radiation monitor is provided with a red beacon to alarm a high-high radiation condition, an amber beacon to alarm a high-radiation condition, and a horn for an audible alarm on either high or high-high radiation. A channel failure alarm light, which is on during normal operation, turns off on detection of channel failure. All alarms are annunciated in the main control room. Alarm set points are adjustable over the range of the detector.

3. Sensitivity - Area monitors are sensitive to gamma radiation of photon energies 100 keV and above.

4. Environmental conditions - The area monitors are designed to operate in the normal environmental conditions for the areas in which they are located for the design life of the plant. The post-accident containment area monitors are designed to remain functional during a DBA.

12.3.4.1.2 Criteria for Location of Area Monitors

Generally, area radiation monitors are provided in areas to which personnel normally have access and for which there is a potential for personnel to receive high-radiation doses (e.g., in excess of 10CFR20 limits) in a short period of time. Plant areas which meet one or more of the following criteria are monitored:

1. Areas which during normal plant operations, including refueling, could exceed radiation limits due to system failure or personnel error.

2. Areas which are continuously occupied following an accident to perform plant shutdown.

3. Areas where new fuel is received and stored. Two detectors are provided at the new fuel storage area of the fuel building to serve as criticality alarms, as specified by 10CFR70.24.

4. Area monitors are located in accordance with the requirements in GDC 63 of 10CFR50, Appendix A.

5. Post-accident containment and drywell monitors are located in accordance with the requirements in GDC 64 of 10CFR50, Appendix A and with the requirements in item II.F.1(3) of NUREG-073

3 6. Post-accident vital area monitors are located in accordance with Regulatory Guide 1.97, Rev. 2, Table 1.

12.3.4.1.3 System Description (Area Radiation Monitoring)

The area radiation monitoring system detects, measures, and records ambient gamma radiation levels at various locations. It also provides audible and visual alarms in monitored areas and in the main control room if gamma radiation exceeds a specified limit. It provides visual indication in

alarms. All monitors are independent, and failure of one monitor has no effect on any others.

The area radiation monitors are powered from the 120-V ac regulated bus. Standby power to this bus is provided by the station battery through an inverter for balance of plant equipment. The redundant post-accident containment monitors are powered by 120-V ac divisional busses.

The location of each area radiation detector is indicated on the radiation zoning and access control drawings, Fig. 12.3-6 through 12.3-10, and is listed in Table 12.3-1. Consistent with the previous criteria, the following general areas are monitored:

1. Areas required for safe shutdown
2. Sampling rooms and health physics laboratory
3. Containment and drywell
4. Fuel storage and handling areas
5. Waste storage and handling areas
6. Maintenance areas.

12.3.4.1.4 Safety Evaluation

The area radiation monitoring system is not essential for the safe shutdown of the plant, and serves only to warn plant personnel of high-radiation levels in various plant areas. Except for the two high-range containment monitors required for post-accident monitoring, the system serves no active emergency function during operation.

The area radiation monitoring system is designed to operate unattended for extended periods of time. A visual display of ambient radiation dose rate and trend information for any detector is available on demand in the main control room. These monitors provide audible and visual alarms at the detector and annunciate in the main control room if the radiation levels exceed preset limits. A strip chart recorder located on the radiation monitor panel in the main control room provides a permanent record of the radiation levels for the two post-accident safety-related containment monitors. Also, analog indication is provided on the main benchboards in the main control room for quick operator assessment of post-accident monitoring (PAM) conditions.

and for the two post-accident safety-related drywell monitors.

TABLE 1A-1 (Cont)

Item and Title	Position	FSAR Reference*	
II.B.1 Reactor coolant system vents	Gulf States Utilities supports the BWR Owners Group position submitted October 17, 1979. Specifically for River Bend Station, primary venting capability is provided by the 16 power operated safety relief valves. Each of the safety relief valves is seismically and Class 1E qualified and the air supply to the seven valves which comprise the automatic depressurization system is seismically qualified. These valves can be manually operated from the main control room to vent the reactor coolant system. Emergency procedures provided to assure core cooling under accident conditions result in system venting and hence no specific venting procedures have been provided. Positive position indication for each valve is provided in the main control room. Additional venting capability is provided via a reactor vessel head vent valve and through operation of the turbine driven reactor core isolation cooling system. No additional accident analyses have been provided as a result of a break in any of these vent lines because a more bounding complete steam line break is part of the River Bend Station design basis.	5.2	3
II.B.2 Plant shielding	The radiation and shielding-design review is currently being performed. Documentation of the review will be provided once the results are finalized. e	12.5 e ← Insert A	7
II.B.3 Post-accident sampling	River Bend Station will have a post-accident sampling system in accordance with NUREG-0737 and related clarification letters.	9.3.2	
II.B.4 Training for mitigating core damage	Training for mitigating core damage will be incorporated into the operator training program. Currently the training program addresses Enclosure 3 to H. R. Denton's March 28, 1980 letter as referenced in NUREG-0737. All other related personnel will receive training commensurate with their responsibilities.	13.2	
II.D.1 Relief and safety valve requirements	Gulf States Utilities participated in a generic test program to satisfy the requirements of TMI Action Plan Item II.D.1. The Crosby SRV used at River Bend Station was included in the test program. The testing requirements were determined by the BWR Owners Group through systematic analysis of design accidents and operational transients. The conclusion from that analysis was "there is no design-basis accident or transient which requires safety, relief, or dual function SRVs to pass two phase or liquid flow at high pressure."	3.9.6B	3

Insert A

Appendix 1A Item II.B.2

A radiation and shielding-design review is being performed on spaces around systems that may require access during and/or after an accident and may contain high radiation levels. In the evaluation of plant shielding and vital area access, post-accident radiation releases equivalent to the source terms described in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant", are assumed. The radiation source terms used in the evaluation are provided in Table 12.2-19.

The results of the evaluation will provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural or administrative controls. The personnel exposure in a vital area will be maintained in accordance with the guidelines in GDC 19 during the course of an accident. The evaluation of plant shielding, vital areas/access and environmental qualification of safety-related equipment which may be used in post-accident situations, will be completed and the documentation submitted in a future amendment to new FSAR Section 12.6 at least four months before the RBS operating license is issued.

This position complies with NUREG-0737 TMI Action Plan Item II.B.2. However, particulates are retained in the suppression pool rather than becoming airborne.

Table 12.2-19
Post Accident
Radiation Source Terms

Initial LOCA Release to containment atmosphere	100% Core Noble Gases 50% Core Iodine
Initial LOCA Release to Reactor Coolant System (Pressurized)	100% Core Noble Gases 50% Core Iodine 1% Core Particulates
Intial LOCA Release to Reactor Coolant System (Depressurized)	50% Core Iodine 1% Core Particulates
High Energy Line Breaks other than LOCA	No core damage
Fuel Handling Accident (from postulated damage of 123 fuel rods)	10% Noble Gases (30% for Kr85, 10% Iodines)

TABLE 1A-1 (Cont)

Item and Title	Position	FSAR Reference*
III.D.1.1 (Cont)	a. Inspecting floor areas and equipment drain cups for wetting which would occur if leakage were present. b. Monitoring the associated equipment or floor drain sump for excessive flow or fill rates.	
III.D.3.3 Inplant Id radiation monitoring	Gulf States Utilities will provide the equipment, procedures and associated training required to accurately determine airborne iodine concentration in areas where plant personnel may be present during an accident. Additional information is provided in revised Section 7.5.	12.3.4 Insert B
III.D.3.4 Control room habitability	Control room habitability requirements are met by the current River Bend Station design.	6.4

7

*FSAR reference section will be updated as necessary in future amendments.

INSERT B

III.D.3.3
Inplant Id Radiation Monitoring

Gulf States Utilities will provide the equipment, procedures and associated training required to accurately determine airborne iodine concentration in areas where plant personnel may be present during an accident. Where stationary monitoring instrumentation is restricted due to its size, or ALARA considerations are present, portable monitoring instrumentation will be used. Under accident conditions, an area will be available to analyze the sample for iodine concentrations. A sufficient number of samplers will be available to sample the vital areas.

Additional information is provided in revised Sections 7.5 and 12.5.

determination of penetrating versus nonpenetrating radiation. TLDs are analyzed on a monthly basis or more frequently if circumstances warrant. TLD readings are normally used as the official record of personnel exposure to radiations. Calibrations and quality control performed on the TLDs and the TLD system are in accordance with NRC requirements.

Direct reading dosimeters are worn by personnel in the controlled area as specified by plant instructions. These dosimeters provide a day-to-day or job-to-job estimate of personnel exposure. Direct reading dosimeters are provided in various ranges and sensitivities as shown in Table 12.5-3.

If a potential for neutron exposure exists, personnel are issued neutron-sensitive TLD badges. These badges normally provide the official record of exposure of personnel to neutron radiation.

In the case of a lost or improper reading on a TLD, the individual exposure is estimated using appropriate methods and documented in accordance with plant procedures.

Extremity dosimetry is provided if extremities are expected to receive exposure significantly higher than that to the body. Its use is normally specified on the RWP. The direct-reading dosimeters are calibrated semiannually, and when damage is suspected, or after repair is performed.

Survey instrumentation for exposure measurement of personnel consist of G-M count rate meters (contamination friskers), portal monitors, and/or hand and foot counters. These instruments are calibrated semiannually or prior to use after undergoing repair. The monitoring instrumentation for personnel listed in Table 12.5-3 is maintained as a minimum.

12.5.2.2.4 Radiation Protection Equipment

Portable air samplers are used to collect samples for laboratory determination of airborne radioactive material concentrations. Air samplers are calibrated for flow semiannually. Samples may be counted for radioactive particulate, radioiodine, and airborne gaseous activities. Portable continuous air monitors are used to monitor airborne concentrations at specific work or field locations. Local indication is provided as well as trend information. Visual and audible alarms are provided with variable set points.

Insert C →

Insert C

The portable air samplers shown in Table 12.5-4 are used to monitor iodine levels in buildings under accident conditions as required in item III.D.3.3 of NUREG-0737. Sample analysis is performed in the Services Building laboratory facilities (hot lab and counting rooms) on instruments described in Section 12.5.2.2.1.

Enclosure 3
RBS FSAR

QUESTION 471.11 (12.2)

As specified in Regulatory Guide 1.70, Section 12.2.2, you should provide a tabulation of the expected airborne radioactivity concentrations in equipment cubicles, corridors and operating areas normally occupied by operating personnel, including sources resulting from reactor vessel head removal and spent fuel operations. In Section 12.2.2, you specified that such areas do not contain airborne sources. Operating experience has shown that equipment leaks have resulted in airborne radioactivity.

RESPONSE

7 The response to this request ~~will be provided by June 1983.~~ is provided in revised Section 12.2.2.

This response and revised Sections 12.3.2.2.2 and 12.5.3.2.1 are consistent with LRG-II positions 2-RAB, 3-RAB and 4-RAB.

varying degrees of activity depending on the detailed system and equipment design.

The radionuclide sources in the liquid radwaste system such as pipes, tanks, filters, demineralizers, and evaporators used in shielding calculations are listed in Table 12.2-12. Information on the solid radwaste system is provided in Section 11.4.

Insert

12.2.2 Airborne Radioactive Material Sources

Equipment cubicles, corridors, and operating areas normally occupied by operating personnel do not contain airborne radioactivity sources. The direction of airflow maintained by the auxiliary, turbine, fuel, and reactor building ventilation systems is from areas of lower potential radioactivity to areas of increasing potential until the air is exhausted. In this manner the normally occupied areas, which do not have sources, can not have airborne radioactivity either. Data from operating BWRs corroborate the general lack of airborne activity in corridors and normally occupied operating areas. Routine air sampling to verify the absence of airborne contamination in corridors and normally occupied operating areas is performed. Radioactive equipment which has the potential for leakage has been installed in separate shielded compartments that are not normally occupied.

For airborne contaminated areas which require occupancy, radiation exposures are estimated as discussed in Section 12.4.1.

12.2.2 Airborne Radioactive Material Sources

Design efforts are directed towards keeping radioactive material contained. Leaks from process systems, refueling and decontamination may lead to airborne radioactivity. Equipment cubicles corridors and areas routinely occupied by operating personnel do not contain significant airborne radioactivity sources. Radioactive equipment which has the potential for leakage is installed in separate shielded compartments that are not routinely occupied. In general, the direction of airflow within the building ventilation systems is from areas of low potential for airborne contamination to areas of increasing potential. In this manner, routinely occupied areas are maintained at low levels of airborne radioactivity. Data from operating BWR's corroborate the general lack of airborne activity in corridors and routinely occupied operating areas (Ref. 2). Air samples and surface contamination swipe samples are performed to verify the absense of airborne and surface contamination.

12.2.2.1 Airborne Sources During Normal Operation

The expected airborne radioactivity concentrations in various plant areas for normal plant operation are shown in Table 12.2-17 and are based on the data given in NUREG-0016[1] and EPRI-495[2]. The ventilation system design is evaluated to determine the distribution of the radioactivity throughout the plant that corresponds to the expected annual releases as specified in Table 11.3-1. However, determination of the airborne concentrations within the containment are based on plant specific analysis that account for the cleanup provided by the containment purge system.

The source of airborne radioactivity within the containment during normal plant operation is derived from the assumed discharge of 2,000 pounds per hour of main steam from the safety relief valves to the suppression pool as well as the drywell leakage of radioactivity evolved from the sumps within the drywell. Cleanup of the containment atmosphere is provided by the 7,000 cfm exhaust from the continuous containment purge system.

Corridors and routine access operating areas within the radwaste building are not expected to have significant airborne radioactivity levels. Equipment cubicles are infrequently accessed and may contain low levels of airborne radioactivity but design provisions are provided to minimize the release of radioactivity.

Radwaste building tanks are filled from the top and as the water splashes into the tanks, dissolved and entrained radioactivity may become airborne. This activity is not released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. Pumps and valves for radioactive systems in the radwaste building are located in separate compartments that are not normally occupied. The radwaste building ventilation design provides airflow from areas of low potential for airborne contamination to areas of increasing potential. This insures that any

leakage from radwaste pumps and valves is not directed into normally occupied areas of the building, but is exhausted from the building. An estimate of the airborne levels expected within radwaste building pump rooms is provided in Table 12.2-17. This is based on a distribution of the airborne radioactivity within radwaste building areas as specified in References 1 and 2 and represents a maximum level to be expected within radwaste building equipment cubicles.

The main potential source of airborne radioactivity within the turbine building is leakage from valves on large lines carrying high pressure steam. The River Bend Station design provides for collection of this leakage and its transported back to the condenser. Therefore, noble gas airborne concentrations are expected to be negligible throughout the turbine building except for within the steam jet air ejector (SJAE) and mechanical vacuum pump (MVP) cubicles. These areas are not occupied during operation and the exhaust from these cubicles is exhausted to the environment after filtration to eliminate the possibility of contamination of adjoining areas.

Airborne activities in the auxiliary building are expected to be low except for the reactor water cleanup (RWCU) pump cubicle. This cubicle is not normally occupied due to radiation levels. However, the exhaust from this cubicle is exhausted to the environment during normal operation and through the standby gas treatment system post-accident.

Radioactive equipment in the Fuel Building which has the potential for leakage is installed in separate shielded compartments which are not normally occupied due to radiation levels. The ventilation in the fuel building directs the air from these cubicles to the environment so that an airborne problem in the normally occupied areas is prevented. The ventilation system is also designed to sweep air from the spent fuel pool surface thereby removing the major portion of the potential airborne contamination. In addition, evaporation from the spent fuel pool is minimized by cooling of the pool.

12.2.2.2 Airborne Sources During Refueling

Experience of operating BWRs has shown that airborne radioactivity can result from the reactor vessel dryer and separator if their surfaces are allowed to dry. Other potential airborne sources could occur during vessel head venting and fuel movement. The airborne radioactive material sources resulting from reactor vessel head removal are minimized by venting prior to removal either to drywell purge exhaust system or to the main condenser, with vacuum supplied by the mechanical vacuum pump. The contribution to the airborne radioactivity due to the reactor vessel internals is minimized by keeping them wet or submerged. Expected airborne concentrations within the containment during refueling are presented in Table 12.2-17 and are based on data given in NUREG-0016.

12.2.2.3 Airborne Sources for Relief Valve Venting

A special consideration for airborne concentrations within the containment is the radioactivity release via relief valve discharge to the suppression pool. The different modes of this type of discharge and the frequency of occurrence are discussed in Chapter 15. The limiting case is the main steam isolation valve closure. This event results in the release of 834,000 lbm of main steam to the suppression pool over a period of approximately three hours. Using the expected source terms for main steam from Table 11.1-1 and accounting for the partitioning of the radioactivity between the air and water, the airborne radioactivity concentrations within the containment are calculated as shown in Table 12.2-18.

12.2 REFERENCES

1. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)", NUREG-0016 Rev. 1, Jan. 1979.

2. "Sources of Radioiodine at Boiling Water Reactors" EPRI NP-495, Research Project 274-1, Final Report Feb. 1978.

TABLE 12.2-17

Expected In-Plant Airborne Radioactivity Concentrations (μCi/cc)

ISOTOPE	CONTAINMENT DURING NORMAL OPERATION	CONTAINMENT DURING REFUELING	RADWASTE BLDG. PUMP ROOMS
KR83M	6.5-11 (1)	NEG.	NEG.
KR85M	3.8-10	NEG.	NEG.
KR85	5.3-10	NEG.	NEG.
KR87	2.1-10	NEG.	NEG.
KR88	7.0-10	NEG.	NEG.
KR89	3.7-12	NEG.	NEG.
XE131M	7.5-11	NEG.	NEG.
XE133M	3.1-10	NEG.	NEG.
XE133	2.0-8	NEG.	NEG.
XE135M	2.3-11	NEG.	NEG.
XE135	3.7-9	NEG.	NEG.
XE137	6.0-12	NEG.	NEG.
XE138	4.5-11	NEG.	NEG.
I131	3.0-11	1.5-10	3.0-11
I132	9.9-12	NEG.	NEG.
I133	5.5-11	2.1-9	4.2-10
I134	9.0-12	NEG.	NEG.
I135	2.0-11	NEG.	NEG.
BR83	1.1-12	NEG.	NEG.
BR84	3.3-13	NEG.	NEG.
BR85	1.5-14	NEG.	NEG.
RB88	2.1-9	NEG.	NEG.
CR51	NEG.	5.5-12	2.0-13
MN54	NEG.	1.5-11	1.1-12
FE59	NEG.	1.6-12	8.5-14
CO58	NEG.	3.6-12	5.6-14
CO60	NEG.	3.3-11	2.0-12
ZN65	NEG.	4.9-11	8.5-14
SR89	NEG.	5.5-13	NEG.
SR90	NEG.	9.8-14	NEG.
ZR95	NEG.	4.4-12	2.3-13
NB95	NEG.	8.7-12	1.1-15
MO99	NEG.	1.6-12	8.5-16
RU103	NEG.	5.2-13	2.8-16
AG110	NEG.	NEG.	NEG.
AG110M	NEG.	3.0-14	NEG.
SB124	NEG.	4.9-13	2.0-14
CS134	NEG.	1.5-11	6.8-13
CS136	NEG.	3.0-12	NEG.
CS137	NEG.	2.6-11	1.1-12
BA140	NEG.	1.2-11	1.1-15
CE141	NEG.	3.6-12	2.0-15
H-3	NEG.	2.0-7	NEG.

(1) $6.5 = 6.5 \times 10^{-11}$

TABLE 12.2-18

Expected Containment Airborne Radioactivity Concentrations
Following MSIV Closure

Concentration (μCi/cc) at Time

ISOTOPE	200 SECS	1000 SECS	5000 SECS	3 HRS	6 HRS	18 HRS
I131	1.3-9[1]	2.8-9	6.7-9	8.3-9	3.0-9	5.2-11
I132	1.8-8	3.7-8	7.5-8	7.6-8	1.1-8	5.3-12
I133	1.7-8	3.6-8	8.5-8	1.0-7	3.4-8	4.2-10
I134	3.7-8	7.1-8	1.1-7	9.4-8	3.1-9	4.2-15
I135	1.7-8	3.5-8	7.9-8	9.1-8	2.4-8	1.3-10
BR83	6.9-11	1.4-10	2.9-10	3.0-10	4.3-11	2.1-14
BR84	6.5-11	1.2-10	1.6-10	1.2-10	9.3-13	NEG.
BR85	2.2-11	1.3-11	8.6-12	5.9-12	NEG.	NEG.
KR83M	9.3-7	1.8-6	3.5-6	3.4-6	4.2-7	9.7-11
KR85M	1.8-6	3.5-6	7.2-6	8.0-6	1.8-6	4.9-9
KR85	4.4-7	4.2-7	3.1-7	2.0-7	7.2-8	1.3-9
KR87	5.5-6	1.1-5	1.9-5	1.7-5	1.2-6	3.8-11
KR88	5.8-6	1.1-5	2.3-5	2.4-5	4.1-6	3.8-9
RB88	9.5-6	1.0-4	5.4-4	7.3-4	4.1-4	2.2-5
KR89	2.4-5	1.5-5	9.9-6	6.8-6	NEG.	NEG.
XE131M	4.5-8	4.6-8	4.6-8	4.0-8	1.4-8	2.6-10
XE133M	2.2-7	3.0-7	4.7-7	5.1-7	1.8-7	2.8-9
XE133	1.2-5	1.4-5	1.7-5	1.7-5	6.0-6	1.0-7
XE135M	6.5-6	9.7-6	9.0-6	6.2-6	8.1-10	6.7-13
XE135	7.8-6	1.4-5	3.0-5	3.4-5	1.0-5	7.3-8
XE137	2.9-5	2.1-5	1.4-5	9.6-6	NEG.	NEG.
XE138	2.1-5	3.3-5	3.2-5	2.2-5	4.7-9	NEG.
CS138	2.2-6	1.9-4	3.6-4	1.4-4	4.2-6	7.7-13

(1) 1.3-9 = 1.3×10^{-9}

QUESTION 471.16 (12.3)

Provide a description of how the airborne radioactivity monitors described in Section 12.3.4 (which consist of noble gas and particulate or only noble gas monitors) can have a minimum capability of detecting 10 mpc-hours of particulate or iodine radioactivity, as specified in NUREG-0800 Section 12.3. You should note that the monitors should be able to detect the 10 mpc-hours from any compartment that has the possibility of containing airborne radioactivity and which normally may be occupied, taking into account dilution in the ventilation system.

RESPONSE

The response to this request ~~will be provided by June 1983.~~ 7

is provided in Section 12.3.4.2.5 and in revised Tables 12.3-1,12.3-2, and 12.3-3.

Power for these monitors is supplied by the 120-V ac regulated bus and 480-V ac MCC.

The radwaste building ventilation exhaust effluent radiation monitors are designed to measure radionuclide releases to be reported and evaluated in accordance with Regulatory Guide 1.21. Since the effluent from general areas of the radwaste building is unfiltered, these monitors also indicate airborne levels of radiation in the building.

Sampling is performed by an automatic isokinetic sampling system with probes and returns located in the exhaust duct. The redundant radwaste building ventilation monitors are similar to the fuel building ventilation exhaust monitors described in Section 12.3.4.2.3.2. High, high-high, and high-high-high radiation levels, channel failure, and sampling system failures are alarmed locally and in the auxiliary control room.

The radwaste building ventilation exhaust monitors are powered by the 120-V ac regulated bus and 480-V ac MCC.

12.3.4.2.4 Safety Evaluation

The in-plant HVAC airborne radioactivity monitors have the safety-related functions of isolating their particular ventilation systems and actuating the associated filtered emergency systems, as discussed in Sections 12.3.4.2.1, 12.3.4.2.2, and 12.3.4.2.3.2 through 12.3.4.2.3.5. These monitors are redundant, Seismic Category I, and powered from the emergency power system.

The combination of the airborne radioactivity monitoring system in conjunction with the administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns through the plant, and plant equipment layout is sufficient to guarantee the safety of personnel throughout all areas of the plant where access is required.

12.3.4.2.5 Sensitivities and Ranges

sufficient sensitivity to detect

Each monitoring system has a minimum detectable concentration such that 10 MPC (maximum permissible concentration as defined in 10CFR20)-hours of airborne particulate and iodine radioactivity can be detected in any compartment which has the possibility of airborne contamination. Monitor sensitivities and ranges are indicated in Table 12.3-2.

The airborne particulate radioactivity monitoring system capabilities, taking into account dilution in the ventilation systems, are summarized in Table 12.3-3. The particulate concentrations corresponding to 1 MPC are developed by normalizing the particulate and iodine (in particulate form) releases from the various buildings as shown in Table 11.3-1. ← Insert

Insert

The minimum detectable concentrations for the airborne radioactivity monitors used at River Bend Station are the following:

1. Particulate 1.5×10^{-13} µCi/cc
2. Iodine 2.6×10^{-12} µCi/cc
3. Gaseous 3.2×10^{-7} µCi/cc

These airborne concentrations are below the MPC levels defined in 10CFR20.

TABLE 12.3-1

AREA DIRECT RADIATION MONITOR LOCATIONS

Equipment Number	Area Monitor	Range (mrem/hr)	Set Point	Monitor Location	
1RMS					1.18
*RE 16A	Primary containment - PAM A	$10^0 - 10^7$ R/hr	See note	Reactor Building El 186' - 3"	1.20 1.21
*RE 16B	Primary containment - PAM B	$10^0 - 10^7$ R/hr	See note	Reactor Building El 186' - 3"	1.24 1.25
*RE 20A	Drywell - PAM A	$10^0 - 10^7$ R/hr	See note	Drywell	1.28
*RE 20B	Drywell - PAM B	$10^0 - 10^7$ R/hr	See note	Drywell	1.31
-RE 138	Personnel airlock to drywell	$10^0 - 10^5$	Upper radiation zone limit	Reactor Building El 130' - 7"	1.34 1.35
-RE 139	Inside annulus near fuel transfer tube	$10^0 - 10^5$	Upper radiation zone limit	Reactor Building El 123' - 6"	1.38 1.39
-RE 140	Refueling floor near north entrance	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 186' - 3"	1.53 1.54
-RE 141	Refueling floor near south entrance	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 186' - 3"	1.57 1.58
-RE 143	RWCU precoat pump and tank area	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 162' - 3"	2.3 2.4
-RE 144	General area	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 141' - 0"	2.7 2.8
-RE 145	Fuel transfer tube area	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 114' - 0"	2.11 2.12
-RE 146	Airlock	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 114' - 0"	2.15 2.16
-RE 147	TIP area	$10^{-1} - 10^4$	Upper radiation zone limit	Reactor Building El 95' - 9"	2.19 2.20

Enclosure 4 (cont'd.)

TABLE 12.3-1 (Cont)

Equipment Number	Area Monitor	Range (mrem/hr)	Set Point	Monitor Location	
-RE 149	Unit cooler general area	10^{-1} - 10^{4}	Upper radiation zone limit	Reactor Building El 162' - 3"	2.27 2.28
-RE 150	Fuel transfer isolation valve area	$2x10^{-2}$ - 10^{7} R/hr	Upper radiation zone limit	Reactor Building El 128' - 1 3/4"	2.31 2.32
-RE 151	Reactor sample station area	10^{-1} - 10^{4}	Upper radiation zone limit	Reactor Building El 162' - 3"	2.35 2.36
-RE 152	Hydraulic control units east bank	10^{-1} - 10^{4}	Upper radiation zone limit	Reactor Building El 114' - 0"	2.39 2.40
-RE 162	Off gas regeneration area	10^{-1} - 10^{4}	Upper radiation zone limit	Off Gas Building El 67' - 6"	2.43 2.44
-RE 164	Off gas sample rack area	10^{-1} - 10^{4}	Upper radiation zone limit	Off Gas Building El 123' - 6"	2.47 2.48
-RE 165	Off gas demineralizer regeneration area	10^{0} - 10^{5}	Upper radiation zone limit	Off Gas Building El 67' - 6"	2.51 2.52
-RE 166	Off gas demineralizer/ strainer valve area	10^{-1} - 10^{4}	Upper radiation zone limit	Off Gas Building El 95' - 0"	2.55 2.56
-RE 167	Off gas building valve area	10^{0} - 10^{5}	Tech. Spec.	Off Gas Building El 137' - 6"	3.1 3.2
-RE 170	Main control room	10^{-2} - 10^{3}	Tech. Spec.	Control Building El 135' - 0"	3.5 3.6
-RE 171	High level counting room	10^{-2} - 10^{3}	Upper radiation zone limit	Services Building	3.9 3.10
-RE 172	Hot machine shop	10^{-2} - 10^{3}	Upper radiation zone limit	Auxiliary Control Building El 95' - 0"	3.13 3.14
-RE 180	Radwaste evaporator distillate cooler area	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 136' - 0"	3.17 3.18
-RE 181	Radwaste sample room	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 106' - 0"	3.21 3.22
-RE 182	Radwaste recovery sample pump area	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 65' - 0"	3.25 3.26

TABLE 12.3-1 (Cont)

Equipment Number	Area Monitor	Range (mrem/hr)	Set Point	Monitor Location	
-RE 183	Radwaste drum storage area east wall	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 106' - 0"	3.29 3.30
-RE 184	Solid waste compactor area	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building EL 106' - 0"	3.33 3.34
-RE 185	Storage tank valve strainer area	10^{0} - 10^{5}	Upper radiation zone limit	Radwaste Building El 90' - 0"	3.37 3.38
-RE 186	Floor drain sump area	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 65' - 0"	3.41 3.42
-RE 187	High conductivity sump area	10^{-1} - 10^{4}	Upper radiation zone limit	Radwaste Building El 65' - 0"	3.45 3.46
-RE 188	Radwaste solidification area	10^{0} - 10^{5}	Upper radiation zone limit	Radwaste Building El 106' - 0"	3.50 3.51
-RE 190	Inside new fuel vault	10^{-1} - 10^{4}	Tech. Spec.	Fuel Building El 113' - 0"	4.6 4.7
-RE 191	Inside new fuel vault	10^{-1} - 10^{4}	Tech. Spec.	Fuel Building El 113' - 0"	4.10 4.11
-RE 192	Fuel building refueling platform	10^{-1} - 10^{4}	Upper radiation zone limit	Fuel Building El 113' - 0"	4.14 4.15
-RE 193	Fuel building operating floor	10^{-1} - 10^{4}	Upper radiation zone limit	Fuel Building El 113' - 0"	4.18 4.19
-RE 194	Fuel transfer tube mid-support room	$2x10^{-3}$ – 10^{7}	Upper radiation zone limit	Fuel Building El 113' - 0"	4.22 4.23
-RE 195	Fuel building sample station	10^{-1} - 10^{4}	Upper radiation zone limit	Fuel Building El 95' - 0"	4.26 4.27
-RE 196	Equipment drain sump area	10^{-1} - 10^{4}	Upper radiation zone limit	Fuel Building El 70' - 0"	4.30 4.31
-RE 200	Turbine building northeast hoist area	10^{-1} - 10^{4}	Upper radiation zone limit	Turbine Building El 123' - 6"	4.35 4.36
-RE 201	Air removal pumps area	10^{-1} - 10^{4}	Upper radiation zone limit	Turbine Building El 95' - 0"	4.39 4.40

TABLE 12.3-1 (Cont)

Equipment Number	Area Monitor	Range (mrem/hr)	Set Point	Monitor Location	
-RE 202	Reactor feed pump area	10^{-1} - 10^{4}	Upper radiation zone limit	Turbine Building El 67' - 6"	4.43 4.44
-RE 203	Turbine building sample room	10^{-1} - 10^{4}	Upper radiation zone limit	Turbine Building El 67' - 6"	4.47 4.48
-RE 204	Condensate demineralizer sample sink area	10^{-1} - 10^{4}	Upper radiation zone limit	Turbine Building El 95' - 0"	4.51 4.52
-RE 210	Remote shutdown panel area	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 114' - 0"	4.55 4.56
-RE 211	Control rod drive maintenance area	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 95' - 9"	5.1 5.2
-RE 212	HPCS area	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 78' - 6"	5.5 5.6
-RE 213	RHR equipment room A	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 70' - 0"	5.9 5.10
-RE 214	RHR equipment room B	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 78' - 6"	5.13 5.14
-RE 215	RHR equipment room C	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 95'- 6"	5.17 5.18
-RE 216	LPCS equipment room	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 78' - 6"	5.21 5.22
-RE 217	HPCS penetration area	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 78' - 0"	5.25 5.26
-RE 218	LPCS penetration area	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 78' - 6"	5.29 5.30
-RE 219	RCIC equipment room	10^{-1} - 10^{4}	Upper radiation zone limit	Auxiliary Building El 95' - 9"	5.34 5.35

Enclosure 4 (cont'd.)

NOTE: Set points for these monitors will be in accordance with Section 13.3, Emergency Planning.

TABLE 12.3-2

AIRBORNE PROCESS AND EFFLUENT RADIATION MONITORS

Equipment Number	QA Cat/ Seismic Category	Monitor Location	Monitor Type	Detector Channels: Channel	Limiting Isotopes	Range µCi/cc	Set Points: High	Alert	Functions	
										1.15 1.16 1.17
1FHS										1.20
*RE 5A	I/I	Fuel Bldg Ventilation Exhaust Fuel Bldg El 148'	Offline Gas Auto-iso-kinetic Sampling	Gaseous	Xe-133	$10^{-7}-10^{5}$	Tech. Spec.		High/alert radiation levels alarm (5A,5B); diverts exhaust air through filter train on high radiation in fuel building exhaust (5A, 5B); Post-accident monitor (5A)	1.2 1. 1.25 1.26 1.27 1.28
*RE 5B	I/I	Fuel Bldg Ventilation Exhaust Fuel Bldg El 148'	Offline Gas & Particulate Auto-iso-kinetic Sampling	Particulate Gaseous	I-131 Xe-133	$10^{-11}-10^{-6}$ $10^{-7}-10^{-2}$ (Particulate iodine sampling to 10^{2} µCi/cc)	Tech. Spec.			1.29 1.30 1.31 1.32 1.33 1.34
-RE 6A	II/Non	Radwaste Bldg Ventilation Exhaust Radwaste Bldg El 166'	Offline Gas Auto-iso-kinetic Sampling	Gaseous	Xe-133 Kr-85	$10^{-7}-10^{5}$ $10^{-7}-10^{5}$	Tech. Spec.		High/alert radiation levels alarm	1.50 1.51 1.52 1.53 1.54 1.55
-RE 6B	II/Non	Radwaste Bldg Ventilation Exhaust Radwaste Bldg El 166'	Offline Gas & Particulate Auto-iso-kinetic Sampling	Particulate Gaseous	I-131 Kr-85 Xe-133	$10^{-11}-10^{-6}$ $10^{-7}-10^{-2}$ $10^{-7}-10^{-2}$ (Particulate iodine sampling to 10^{2}µCi/cc)	Tech. Spec.		High/alert radiation levels alarm	1.57 1.58 2.1 2.2 2.3 2.4 2.5
*RE 7A	I/I	Main Plant Exhaust Duct Aux. Bldg El 170'	Offline Gas Auto-iso-kinetic Sampling	Gaseous Gaseous	Xe-133 Kr-85	$10^{-7}-10^{5}$ $10^{-7}-10^{5}$	Tech. Spec.		High/alert radiation levels alarm (7A, 7B); Post-accident monitor (7A)	2.7 2.8 2.9 2.10 2.11 2.12

Enclosure 4 (cont'd.)

TABLE 12.3-2 (Cont)

Equipment Number	QA Cat/ Seismic Category	Monitor Location	Monitor Type	Detector Channels: Channel	Detector Channels: Limiting Isotopes	Detector Channels: Range µCi/cc	Set Points: High	Set Points: Alert	Functions
1RMS									
*RE 7B	II/Non	Main Plant Exhaust Duct Aux Bldg El 170'	Offline Gas & Particulate Auto-iso-kinetic Sampling	Particulate Gaseous	I-131 Kr-85 Xe-133	10^{-11}-10^{-5} 10^{-7}-10^{-2} 10^{-7}-10^{-2} (Particulate iodine Sampling to 10^{-2} µCi/cc)	Tech. Spec.		
*RE 111	I/I	Primary Containment Atmosphere Containment El 162'	Offline Gas & Particulate Sampling Trees	Particulate Gaseous	I-131 Xe-133 Kr-85	10^{-11}-10^{-5} 10^{-7}-10^{-1} 10^{-7}-10^{-1}	Tech. Spec.		High/alert radiation levels alarm; Reactor coolant pressure boundary leak detection
*RE 11A *RE 11B	I/I	Reactor Bldg Annulus Ventilation Aux Bldg El 170' El 141'	Offline Gas Isokinetic Sampling	Gaseous	Xe-133	10^{-6}-10^{-1}	Tech. Spec.		High/alert radiation levels alarm; Initiates standby gas treatment system on high radiation
*RE 112	I/I	Drywell Atmosphere Containment El 141'	Offline Gas & Particulate Sampling Trees	Particulate Gaseous	I-131 Xe-133 Kr-85	10^{-11}-10^{-5} 10^{-7}-10^{-1} 10^{-7}-10^{-1}	Tech. Spec.		High/alert radiation levels alarm; Reactor coolant pressure boundar leak detection
*RE 13A *RE 13B	I/I	Main Control Room Local Intake Control Bldg	Offline Gas Isokinetic Sampling	Gaseous	Xe-133 Kr-85	10^{-6}-10^{-1} 10^{-6}-10^{-1}	Tech. Spec.		High/alert level alarms; Diverts exhaust air through filter train on high radiation of local outside air intake
*RE 14A *RE 14B	I/I	Main Control Room Remote Intake	Offline Gas Isokinetic Sampling	Gaseous	Xe-133 Kr-85	10^{-6}-10^{-1} 10^{-6}-10^{-1}	Tech. Spec.		High/alert radiation levels alarm

TABLE 12.3-2 (Cont)

Equipment Number	QA Cat/ Seismic Category	Monitor Location	Monitor Type	Detector Channels: Channel	Detector Channels: Limiting Isotopes	Detector Channels: Range $\mu Ci/cc$	Set Points: High	Set Points: Alert	Functions	
1E5S										
-RE 110	II/Non	Auxiliary Bldg. Ventilation Aux Bldg El 141'	Offline Gas & Particulate Auto-isokinetic Sampling	Particulate Gaseous	I-131 Xe-133 Kr-85	10^{-11}-10^{-5} 10^{-7}-10^{-1} 10^{-7}-10^{-1}	Tech.	Spec.	High/alert radiation levels alarm	3.6 3.7 3.8 3.9 3.10
-RE 118	II/Non	Turbine Bldg Ventilation Turbine Bldg El 123'	Offline Gas & Particulate Auto-isokinetic Sampling	Particulate Gaseous	I-131 Xe-133 Kr-85	10^{-11}-10^{-5} 10^{-7}-10^{-1} 10^{-7}-10^{-1}	Tech.	Spec.	High/alert radiation levels alarm	3.12 3.13 3.14 3.15
-RE 124	II/Non	Cond/Demin & Off Gas Bldg Ventilation Turbine Bldg El 123'	Offline Gas & Particulate Auto-isokinetic Sampling	Particulate Gaseous	I-131 Xe-133	10^{-11}-10^{-5} 10^{-7}-10^{-1} 10^{-7}-10^{-1}	Tech.	Spec.	High/alert radiation levels alarm	3.18 3.19 3.20 3.21 3.22

Enclosure 4 (cont'd.)

TABLE 12.3-3

AIRBORNE PARTICULATE RADIOACTIVITY MONITORING CAPABILITIES

Equipment Number	Monitored Area	Ventilation Flow (cfm)	Particulate Concentration* (uCi/cc)	Monitor Sensitivity 10-Hr Buildup (uCi/cc)
RMS				
RE 118	Total turbine building	44,000	2.7 E-8	10^{-12}
	Turbine building area with minimum ventilation	200 (MVP area)	1.2 E-10	10^{-12}
RE 9A	Total containment building	7,000	2.8 E-8	10^{-12}
	Containment area with minimum ventilation	500 (RWCU pump cubicle)	6.0 E-10	10^{-12}
RE 5B	Total fuel building	10,000	2.8 E-8	10^{-12}
	Fuel building area with minimum ventilation	150 (SFC demineralizer cubicle)	4.1 E-10	10^{-12}
RE 110	Total auxiliary building	10,000	3.3 E-8	10^{-12}
	Auxiliary building area with minimum ventilation	200 (CRD work area)	6.6 E-10	10^{-12}
RE 124	Total offgas building	9610	8.7 E-8	10^{-12}
	Offgas building area with minimum ventilation	100 (Prefilter area)	9.0 E-10	10^{-12}
RE 6B	Total radwaste building	80,400	1.2 E-8	10^{-12}
	Radwaste building area with minimum ventilation	730 (Radwaste filter and demineralizer area)	1.1 E-10	10^{-12}

*This concentration corresponds to 1 MPC of airborne particulate radioactivity in the monitored area taking into account dilution in the ventilation system.

Enclosure 4 (cont'd.)

QUESTION 471.10 (12.4)(12.1)

In Section 12.4.1 and 12.1.2.5, you stated that the occupational dose assessment will be provided in a later amendment. Regulatory Guide 1.70, Section 12.4, specifies that you should perform such a dose assessment. You should provide a copy of this assessment in accordance with Regulatory Guide 8.19 and a listing of plant improvements you will make as a result of this review.

RESPONSE

The response to this request is provided in revised Sections 12.1.2.5 and 12.4.1. This dose assessment will include exposure from SRV actuation consistent with LRG-II position 1-RAB.

12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily from components and equipment containing radioactive fluids, and to a lesser extent from the presence of airborne radionuclides. In-plant radiation exposures during normal operation and anticipated operational occurrences are discussed in Section 12.4.1. Radiation exposures at other onsite locations outside the plant which arise from onsite radioactive sources, the presence of N-16 in the plant, and radioactive gaseous effluents are discussed in Section 12.4.2.

Radiation exposure to construction workers in Unit 2 construction area is due primarily to the presence of N-16 inside the operating Unit 1 buildings. These exposures will be evaluated pending finalization of the Unit 2 construction schedule.

Radiation exposures to operating personnel are within 10CFR20 limits. Radiation protection design features described in Section 12.3 and the health physics program outlined in Section 12.5 assure that the occupational radiation exposures (ORE) to operating and construction personnel during operation and anticipated operational occurrences are as low as is reasonably achievable (ALARA).

12.4.1 Exposures Within the Plant

The occupational dose assessment in accordance with Regulatory Guide 8.19 is in progress as part of the River Bend Station ALARA design review. The dose assessment and a listing of any resulting plant improvements will be provided in a future amendment. 3

Insert

12.4.2 Exposures at Locations Outside the Plant Structures

Radiation exposures at locations outside the plant arise from: 1) onsite radioactive sources outside plant buildings, 2) direct and air-scatter (skyshine) contributions due to the presence of N-16 in the plant buildings, and 3) release of gaseous effluents from the plant. The dose due to N-16 is the predominant contributor. Estimated doses for the restricted area boundary are summarized in Table 12.4-2. These estimates meet the dose guidelines of 10CFR20 and 40CFR190.

LRG-II Issue 1-RAB

INSERT:

RBS will estimate personnel exposure resulting from the actuation of SRV's based on Reference 2. The dose analysis for the standard plant design in the report is applicable to RBS because of the similarity of designs.

The safety relief valve discharge event considered in the analysis is the Type 2 isolation event, in which the reactor pressure is initially controlled by the cyclic lifting of the SRV's. All SRV's are assumed to open with the low set relief valves remaining open following the closure of the other valves. Design basis radiation sources for normal operation are used in the analysis.

Normal ventilation in containment is assumed and airborne concentrations are not corrected for plateout on the walls. The dose assessment methodology including the pool retention factors are provided in the reference document.

Assuming a conservative egress time of four minutes for an operator located in the TIP drive area, the estimated doses for RBS personnel exiting containment following a Type 2 blowdown event will be provided in a future amendment to the FSAR.

Reference - 12.4

1. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants, RP-8A. Stone & Webster Engineering Corporation, Boston, MA, May 1975.

2. General Electric Company document 22A5718, Revision 1, "Mark III Containment Dose Reduction Study", January 29, 1980.

Enclosure 6

RBS FSAR

QUESTION 471.21 (13.1)

As specified in Regulatory Guide 1.70, Section 13.1.1.ᴊ and NUREG-0737, you should provide an outline of the qualifications of the individuals designated as your Radiation Protection Supervisor (RPS) and his backup. It is our position that RPS have the qualifications specified in Regulatory Guide 1.8 for Radiation Protection Manager. The December 1979 revision of ANSI 3.1 specifies that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, two years experience in radiation protection, one year of which should be nuclear power plant experience, six months of which should be on-site. It is our position that such experience should be professional experience. Identify and provide an outline of the qualification of the individual who will act as the backup for the RPM in his absence.

RESPONSE

~~Qualifications of the Radiation Protection Supervisor and the normal backup will be provided as stated in Section 13.1.3.2.~~ Individuals temporarily filling the Radiation Protection Supervisor position are required to have a BS degree in science or engineering (or the equivalent), 2 years of experience in radiation protection, 1 year of which is in nuclear power plant experience, 6 months of which would be onsite. 5

The recent reorganization has added the position of the Radiation Protection/Chemistry Supervisor who will have the qualifications specified in Regulatory Guide 1.8 for the Radiation Protection Manager. The Radiation Protection Supervisor acts as his normal backup.

13.1.2.2.11 Radiation Protection/Chemistry Supervisor

The Radiation Protection/Chemistry Supervisor is responsible for the management of the RBS radiation protection program, the direction of all radiation protection department personnel, and directing the sampling and analysis of Plant Fluid Systems as well as evaluating and reporting the results. He supervises the radiation, environmental, and personnel monitoring programs, the ALARA program, the respiratory protection program, and the whole body counting program. He ensures that adequate radiation protection training has been given to all plant staff and emergency team members and that they have completed training and medical qualifications prior to working in radiation areas.

The Radiation Protection/Chemistry Supervisor reports to the Assistant Plant Manager, Operations, but has direct access to the Plant Manager on all radiation protection matters. The normal backup to the Radiation Protection/Chemistry Supervisor is the Radiation Protection Supervisor. The qualifications of this individual are provided in Section 13.1.3.2.

in the annulus is plugged with a 5 1/2-ft-thick, solid concrete block to maintain the radiation levels in the area within the zoning requirements. The seismic shake spaces between the steel containment and the structures inside and outside the containment are shielded by steel plates. Fig. 12.3-11 shows the reactor building areas through which the spent fuel transfer tube passes.

Insert A →

Turbine Building

The anticipated major radiation source in the turbine building is the primary steam containing activation gases, principally N-16, and fission products. Radiation shielding is provided around the following equipment in order to ensure that the required access zone radiation limits are met around the shielded areas:

1. Main condenser hot well
2. Feedwater heaters and drain receiver tanks
3. Steam air ejectors
4. Steam extraction piping
5. Off gas equipment and piping
6. Condensate demineralizers and regeneration facilities
7. Steam seal evaporator
8. Turbines
9. Radwaste steam reboiler
10. Moisture separator/heaters.

Areas within these shields are high-radiation zones and have limited access.

Auxiliary and Radwaste Buildings

Concrete walls, removable blocks, labyrinths, and pipe chases are used to shield the safety feature equipment in the auxiliary building and process equipment in the radwaste building, including valves and piping, in accordance with general guides of Section 12.3.2.2.1. Radiation limits are specified for all access zones, and shielding is provided as necessary to maintain the zones within these limits.

LRG-II Issue 2-RAB

INSERT A:

RBS radiation zone maps identify the maximum expected total radiation levels during operations and refueling with consideration given toward maintaining personnel exposure as low as is reasonably achievable and within the standards of 10CFR20. The zone maps show all the routinely visited areas such as reactor water cleanup (Figure 12.3-4, el 162'-3"), standby liquid control areas (Figure 12.3-3, el 141"-0"), TIP Station (Figure 12.3-2, el 98"-0"), CRD hydraulic control unit (Figure 12.3-4, el 114"-0") and containment personnel airlocks (Figure 12.3-4, el 114'-0" and el 162'-3"). These zone radiation levels include contributions from any potential streaming through the drywell shield wall penetration.

12.5.3.2.1 Refueling

Procedures and methods used to maintain radiation exposure ALARA during refueling outages include the following:

1. An RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures and equipment to be used.

3. Before removing the vessel head, the primary system is degassed and sampled to minimize expected airborne activity when the head is removed.

4. During movement of irradiated fuel assemblies, the active fuel is maintained under at least 8 ft 6 in of water.

5. The refueling cavity water is filtered to reduce the activity in the water and to lower exposure rates.

6. Radiation levels in work areas are monitored and precautions taken as necessary, consistent with ALARA.

7. Filtered or exhaust ventilation is operated as appropriate to minimize airborne radioactive material.

Insert B

12.5.3.2.2 Inservice Inspection

Protection procedures and methods used during inservice inspections to maintain radiation exposure ALARA are as follows:

1. An RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures, equipment, radiation and contamination levels, and protective clothing requirements appropriate to a particular job.

3. Insulation is designed, where practical, for ease of removal and replacement where removal is required for repetitive inspections.

ENCLOSURE 7 (Cont'd)

LRG-II Issue 3-RAB

INSERT B:

To minimize personnel exposure from potentially high airborne radioactivity concentrations during refueling, equipment is maintained wet during the short time the steam dryer and part of the steam separator are out of water. In addition, administrative controls are implemented to minimize personnel exposure using respiratory protection equipment when necessary and containment access control during the transfer operation.

12.3.2.2.2 Plant Shielding Description

Plant building layouts, which provide locations of equipment containing radioactive fluids and indicate shield wall thickness, radiation zone designations, access control, and radiation monitor locations, are shown in Fig. 12.3-6 through 12.3-10.

The general description of plant shielding in the different plant buildings is as follows.

Reactor Building

Shielding for the reactor building includes the primary shield, drywell, and shield building walls.

The primary shield wall surrounds the reactor and reduces gamma heating in the drywell concrete wall; reduces activation of, and radiation effects on, materials and equipment in the drywell; and provides limited access in the drywell for shutdown periodic inspection and maintenance. The drywell is Zone VI during normal operation.

The drywell wall provides additional shielding in order to permit access to the containment during normal operation. The open containment radiation level in most areas outside the drywell is less than 2 mrem/hr.

Within the containment, there are several shielded rooms. The rooms enclose reactor water cleanup system equipment and piping, and safeguard and process equipment to maintain radiation levels in the containment at less than 2 mrem/hr. In addition, the main steam lines are within a shielded tunnel.

The shield building wall is a 2 1/2-ft-thick, concrete structure that completely surrounds the nuclear steam supply system. This wall attenuates the system radiation to ensure that levels outside the building are less than 0.2 mrem/hr. In addition, in the unlikely event of an accident, the shield building wall shields personnel and the public from radiation sources inside the containment.

Details on the construction of these walls are presented in Section 3.8.

Insert C →

Access to the fuel transfer tube area within the reactor building is through a 2 1/2 x 2 1/2 ft opening that is located in the annulus and through an entrance to the isolation valve room in the containment. The access opening

ENCLOSURE 7 (Cont'd)

LRG-II Issue 4-RAB

INSERT C

RBS provides shielding designs for the fuel transfer tube and canal commensurate with the guidance of Regulatory Guide 8.8 that results in radiation doses within the limits of 10CFR20.

Access to areas where contact with the fuel transfer tube may occur is administratively controlled. Radiation monitors with audible and visible alarms are provided for these areas. Further protection against inadvertent personnel exposures is assured through system interlocks that prevent fuel transfer tube operation when these accessible areas are unsecured. Signs are posted stating that potentially lethal radiation fields are possible during fuel transfer.