Georgia Power Company
Post Office Box 282
Waynesboro, Georgia 30830
Telephone 404 554-9961
404 724-8114

Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 870-6011

Vogtle Project

May 5, 1988

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

File: X7BC35
Log: GN-1448

PLANT VOGTLE - UNIT 2
NRC DOCKET NUMBER 50-425
CONSTRUCTION PERMIT NUMBER CPPR-109
SPENT FUEL RACKS

Gentlemen:

NRC letter dated April 7, 1988, transmitted questions on the Vogtle Unit 2 Spent Fuel Racks. Georgia Power Company's (GPC) response to your questions is enclosed. These responses address your questions on occupational radiation exposure, radioactive wastes, accident analyses, potential releases of radioactive materials, offsite radiological impacts, and the boraflex material being utilized.

If your staff requires further information, please do not hesitate to contact me. If necessary, GPC will meet with the NRC staff to explain our responses or address any other questions so that your review can be completed on schedule.

Sincerely,

J. A. Bailey

J. A. Bailey
Project Licensing Manager

JAB/wkl
Enclosure

xc: NRC Regional Administrator
NRC Resident Inspector
P. D. Rice
L. T. Gucwa
R. A. Thomas
B. W. Churchill, Esquire
J. B. Hopkins (2)
G. Bockhold, Jr.
R. J. Goddard, Esquire
R. W. McManus
Vogtle Project File

B001
1/1

RPB #1 Provide the following information:

a. Sources in the Spent Fuel Pool Water

Provide a description of fission and corrosion product sources in the spent fuel pool (SFP) water from: (a) introduction of primary coolant into SFP water, (b) movement of fuel from the core into the pool, and (c) defective fuel stored in the pool. Include a listing of the radionuclides and their concentrations (expressed in mCi/mL) expected during normal operations and refueling. The radionuclides of interest should include ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs.

Response

Fission and corrosion product sources in the spent fuel pool water from (a) and (b) are shown in FSAR Table 12.2.1-19 for shield wall design. Because of its higher concentration, the introduction of primary coolant is the major contributor to SFP water radionuclide concentration. The more dense storage of spent fuel will not have any impact on the contribution of the SFP water concentration from the introduction of primary coolant.

The only contributor to the SFP water radionuclide concentration that could be impacted by the more dense storage of fuel is from (c), additional older defective fuel stored in the pool. Leakage from additional older defective fuel is not expected to increase the spent fuel pool radionuclide concentration; first, because defective fuel is not the major contributor to SFP water radionuclide concentrations (see above) and, second, because the SFP purification system will be used to maintain the radionuclide concentration at an acceptable level.

As discussed in footnote (a) to FSAR Table 12.2.1-19 and subsection 9.1.3.5, the dominant gamma-emitting isotopes in the spent fuel pool water are controlled to maintain the dose rate at the pool surface to 2.5 mrem/hr or less. These pages of the FSAR are attached for your convenience.

b. Airborne Radioactive Sources

Provide a description of radioactive materials that may become airborne as a result of failed fuel and evaporation (e.g., ^{85}Kr, and ^{3}H, respectively). The radionuclide description should include calculated or measured concentrations expected during normal operations and during refuelings.

Response

FSAR Table 12.2.2-2 (attached for your convenience) provides the airborne concentration of radionuclides from the spent fuel pool. As discussed in Table 12.2.2-1 the partition factors for noble gases, halogens and particulates are negligible. Only tritium may be present in a detectable quantity during refueling. The reactor coolant system is the major contributor to airborne radioactive sources from the spent fuel pool. The more dense storage of spent fuel will not have any impact on the contribution to the SFP airborne concentration from the introduction of primary coolant.

No significant increase in the airborne radionuclide concentrations are expected to occur from the more dense spent fuel storage. Leakage from defective fuel is not expected to increase the airborne concentration because defective fuel is not a major contributor to the SFP airborne radionuclide concentrations and because the SFP purification system will be used to maintain the evaporating radionuclide concentration (in the SFP water) at a low level.

c. Miscellaneous Sources of Exposure

Address the effects of more frequent replacement of demineralizer filters on cumulative dose equivalent if this is a factor that results from the modification.

Response

As discussed in (a) above the increase in spent fuel pool storage locations and resultant increase in defective fuel assemblies stored in the pool is not expected to increase the spent fuel pool water radionuclide concentrations. Should an increase in spent fuel pool water radionuclide concentrations occur, the SFP purification system will be used to reduce the concentration to acceptable levels.

Demineralizer resin bed changeout and filter backflush operations are performed remotely from low radiation areas. Control panels and valve reach rods are located in areas designed to maintain radiation levels of 2.5 mr/hr or less, and expected radiation levels are much less. Based on the design activities for the spent fuel pool filter and demineralizer compared to other demineralizers and filters as discussed in FSAR Chapter 12, an increase in SFP purification system resin changeout or filter backflush frequency will have a negligible impact on the activity processed by the solid waste system. Therefore, the increased storage capacity will have negligible impact on plant cumulative doses.

RPB #2

Dose Rates from Fuel Assemblies, Control Rods, and Burnable Poison Rods

a. Provide a description of the dose rate at the surface of the pool water from the fuel assemblies, control rods, burnable poison rods or any miscellaneous materials that may be stored in the pool. Additionally, provide the dose rate from individual fuel assemblies as they are being placed into the fuel racks. Information relevant to the depth of water shielding the fuel assemblies as they are being transferred into the racks should be specified. If the depth of water shielding over a fuel assembly while it is being transferred to a spent fuel rack is less than 10 feet, or the dose rate 3 feet above the spent fuel pool (SFP) water is greater than 5 mR/hr above ambient radiation levels, then submit a Technical Specification specifying the minimum depth of water shielding over the fuel assembly as it is being transferred to the fuel rack and the measures that will be taken to assure that this minimum depth will not be degraded.

Response

The dose rate at the surface of the pool water from the fuel assemblies, control rods, burnable poison rods or any miscellaneous materials that may be stored in the spent fuel pool fuel racks is conservatively estimated as less than 0.05 mR/hr. When fuel assemblies are being placed into the fuel racks the dose rate at the surface of the pool water is conservatively estimated as less than 2.5 mR/hr. This radiation dose rate occurs when the fuel handling machine has lifted the fuel assembly to the upper limit of travel, which together with water level control, results in the maintenance of a minimum water cover of at least 10'-0" over the top of the active fuel. As discussed in FSAR subsection 9.1.4.3.4 (attached for your convenience) this will maintain the gamma dose rate at the surface of the water at 2.5 mrem/hr or less. Therefore, a Technical Specification specifying a minimum water depth over the assembly being transferred is not warranted.

b. Address the dose rate changes at the sides of the pool concrete shield walls, where occupied areas are adjacent to these walls, as a result of the modification. Increasing the capacity of the pool may cause spent fuel assemblies to be relocated close to the concrete walls of the pool, resulting in an increase of radiation levels in occupied areas. Please evaluate this potential problem.

Response

The radiation dose rates around the outside of the pool would increase locally should freshly discharged fuel be located in the cells adjacent to the SFP liner. The dose rates on level B of the fuel handling building would be approximately 17 mr/hr along the west and south SFP walls and approximately 325 mr/hr along the east SFP wall if freshly discharged fuel is located next to the walls. The dose rates would decrease to below 2.5mr/hr after approximately 2 months and approximately 19 months, respectively. The other occupied areas in the fuel handling building would remain less than 2.5 mr/hr from the stored spent fuel assemblies.

During transfer of the spent fuel assembly into its storage location, the dose rates on level A and the operating deck near the gates could locally increase if the fuel assembly is being moved into a storage location adjacent to the SFP wall. The localized dose rate on the level A corridor could be 17 mr/hr when transferring freshly discharged fuel into cell locations adjacent to the south SFP wall. If freshly discharged spent fuel is being transferred into locations near the cask loading pit gate and the cask loading pit canal is dry, the dose rate on the operating deck in the vicinity of the cask loading pit will be administratively controlled to maintain 2.5 mr/hr or less on the operating deck.

The temporary increase in dose levels adjacent to the SFP walls will require Health Physics to control access to these areas and/or operations to permit the spent fuel to decay sufficiently to maintain occupied areas of the fuel handling building at 2.5 mr/hr or less from the spent fuel assemblies.

RPB #3

Dose Rates from SFP Water

Provide information on the dose rates at the surface of SFP water resulting from radioactivity in the water. Include: (1) dose rate levels in occupied areas and along the edges and center of the pool and on the fuel handling crane; (2) effects of crud buildup; and (3) based on refueling water activity, the dose rates before, during, and after refueling.

Response

As discussed in FSAR subsection 9.1.3.5 (attached for your convenience) the dose rate from radioactivity in the SFP water on the fuel handling machine and along the edges of the pool are expected to be 2.5 mrem/hr or less. As discussed in response to RPB #1, the increased storage capacity is not expected to significantly increase the activity in the SFP water. The SFP purification system is used to maintain SFP water quality, prevent a buildup of crud in the SFP, and maintain the dose rates due to dominant gamma-emitting isotopes to 2.5 mrem/hr or less before, during, and after refueling.

RPB #4

Dose Rates from Airborne Isotopes

Based on the source terms, provide the dose rates from submersion and dose commitments from exposure to the concentration of ^{85}Kr and ^{3}H.

Response

As discussed in response to question RPB #1 b., the airborne radionuclide concentrations are provided in FSAR Table 12.2.2-2.

The airborne radioactivity dose estimates are discussed in FSAR paragraph 12.4.1.2 and Table 12.4.1-14 (attached for your convenience). Only ^{3}H is expected to be airborne in detectable quantities. The concentrations shown in Table 12.2.2.-2 in the Fuel Handling Building (2.50E-6μci/cc) represents 50% of the maximum permissable concentration for restricted areas as shown in 10CFR20, Appendix B.

RPB #5

Dose Assessment from Modification Procedures

a. Discuss the manner in which occupational exposure will be kept ALARA during the modification. Include the need for and the manner in which cleaning of the crud on the SFP walls will be performed to reduce exposure rates in the SFP area.

b. Discuss vacuum cleaning of SFP floors if divers are used and the distribution of existing spent fuel stored in racks to allow maximum water shielding to reduce dose rates to divers.

c. Describe plans for cleanup of the SFP water to minimize radioactive contamination and to ensure fuel pool clarity and underwater lighting acceptance criteria to help ensure good visibility.

d. Discuss underwater radiation surveys that will be made before any diving operation. These surveys should be performed before or after any fuel movements or movements of any irradiated components stored in the pool.

e. State your intent to equip each diver with a calibrated alarming dosimeter and personnel monitoring dosimeters, which should be checked periodically to ensure that prescribed dose limits are not being exceeded.

f. Discuss any preplanning of work by divers as required.

g. Discuss your provision for surveillance and monitoring of the spent fuel pool work area by Health Physics personnel during the modification.

Response

Georgia Power Company will be installing twenty (20) free standing racks in the Unit 2 pool at Vogtle. This pool is vacant and has never contained racks or fuel. A steel liner plate covers the pool floor and walls. GPC plans to install, position, and level all twenty (20) racks prior to storage of spent fuel in the Unit 2 pool. Thus, the Vogtle job does not constitute a "Re-rack". The radiological hazards of a "re-rack" job will not apply to Vogtle.

RPB #6

Provide an estimate of the total man-rem to be received by personnel occupying the spent fuel pool area based on all operations in that area including those resulting from (2), (3), and (5) above. Describe the impact of the spent fuel storage rack modification on these estimates.

Response

Total man-rem estimates for the plant based on all operations including refueling are discussed in FSAR section 12.4. As discussed in the responses to (2), (3), and (5) above, the additional spent fuel storage capacity and the installation of this capacity is not expected to result in increases of the radiation levels in the normally occupied areas of the fuel handling building. Appropriate administrative controls will be applied to operations in the fuel handling building to maintain radiation levels consistent with plant radiation zoning and access control as described in FSAR subsection 12.3.1.2 (attached for your convenience). Therefore, the additional spent fuel storage capacity is not expected to impact these man-rem estimates.

CHEB #1

Based on the recent experience pertaining to degradation of Boraflex in spent fuel pools at Quad Cities and Point Beach nuclear power plants, provide justification to demonstrate the continued acceptability of Boraflex for application in the Vogtle spent fuel pool.

Response

Transmission measurements and neutron radiography in the Quad Cities spent fuel storage racks confirmed the existence of a number of gaps in the Boraflex, distributed randomly in both size and axial location (above the lower 4 feet) with the largest gap (as determined by radiography) being approximately 3.5 inches in width corresponding to approximately 2.5% shrinkage in length. These gaps have been attributed to the rack manufacturing process which rigidly clamped the Boraflex in a manner that did not allow the Boraflex to shrink unrestrained. (A k-effective analysis of the Quad Cities spent fuel storage pool demonstrated that these gaps do not cause the Quad Cities racks to exceed the 0.95 limit on k-effective.

Two full length panels of Boraflex were removed from the Point Beach racks after small surveillance samples showed evidence of degradation. Both of the full length panels (one unirradiated and one with a twenty-year equivalent exposure of approximately 1.6×10^{10} rads) were intact and capable of performing their design function. These measurements confirmed that, although some radiation induced changes in physical properties had occurred, the Boraflex retained its neutron absorbing properties and will therefore, continue to assure criticality safety.

Earlier irradiation tests of Boraflex showed a negligible loss of boron at irradiation levels up to 1×10^{10} rads gamma (or approximately 5×10^{12} rads total including the concurrent neutron exposure in the test reactor). Subsequent tests (1) confirmed that Boraflex retains its neutron absorbing properties (i.e., boron is not lost on irradiation) in irradiations equivalent to the expected inservice lifetime of the racks. Above an irradiation level of approximately 1×10^{9} rads, Boraflex becomes a hard ceramic-like material which remains stable over irradiations comparable to a 40-year service life in the spent fuel pool. Shrinkage approaches a level of 2 to 2-1/2 percent in length with a slightly greater shrinkage in width observed, probably due to a small amount of edge deterioration.

(1) Irradiation Study of Boraflex Neutron Absorber, Interim Test Data, Bisco Products, Inc., Technical Report NS-1-050, Novmeber 1987; (attached for your convenience).

The design of the Vogtle spent fuel racks and the manufacturing process specifically incorporates measures to allow unrestrained shrinkage of the Boraflex, and thereby preclude any mechanism that would cause gaps to be produced. The Boraflex sheets are initially oversized to provide a 3 inch allowance (2.1%) for shrinkage. In the manufacturing process, no adhesives are used and the Boraflex sheets are carefully installed in a non-stretched condition and without any tears or cracks. Thus there is reasonable assurance that the Boraflex in the Vogtle racks will continue to be acceptable in the Vogtle spent fuel pool for the expected service lifetime of the racks and maintain k-effective so as not to exceed the 0.95 limit.

CHEB #2

Based on the recent information, provide any changes, to the inservice surveillance program for Boraflex neutron absorbing material and describe the frequency of examination and acceptance criteria for continued use. Provide procedures for testing the Boraflex material and interpretation of test data.

Response

Since Vogtle Unit 2 currently contains no poisoned fuel racks, a surveillance program for the Boraflex neutron absorbing material is still in the developmental process. This surveillance will monitor changes in the Boraflex sample coupons as follows:

Physical Characteristics:

a) Visual examination to determine changes in the color, texture, or shape or whether pitting, cracking, or similar phenomena has occurred.

b) Detailed dimensional examination.

c) Measurement of specific gravity.

Nuclear Characteristics:

a) Neutron attenuation measurement to determine B-10 concentration.

b) Neutron radiograph to determine uniformity of boron distribution.

Where appropriate, physical characteristics will be determined in accordance with applicable ASTM testing methods. Test data will be evaluated based on current BISCO (the manufacturer of Boraflex) guidelines. Acceptance criteria will be based on shrinkage and the ability of the racks to maintain k-effective $\leq$ 0.95. Coupon testing will be initially performed at regular intervals, based on refueling cycles. This may be modified based on test data, and industry and EPRI recommendations. The surveillance program will be developed prior to the use of the racks for storage of spent fuel.

The surveillance program will be sufficient to detect any significant changes in the neutron attenuation properties of the Boraflex or any changes in the physical structure which may be indicative of possible distribution anomolies of the Boraflex. As stated in question CHEB #1, the racks are designed to accommodate shrinkage, and the surveillance program will monitor this parameter. As a result, this surveillance program will assure that the Boraflex in the spent fuel racks will be acceptable for continued use.

CHEB #3

Describe the corrective actions to be taken if degraded Boraflex specimens or absorber is found in the spent fuel pool.

Response

As discussed in the responses to questions CHEB #1 and #2, it is expected that Boraflex will perform its design function throughout the lifetime of the spent fuel racks. The effects of the spent fuel environment on the Boraflex have been incorporated into the design, and the Boraflex surveillance program will be designed to provide assurance that the Boraflex is performing as expected. In the event of unexpected detection of unacceptable degradation of the Boraflex samples and subsequent indication that the Boraflex in the spent fuel storage cells might become unable to perform its design function, there are a number of remedial steps available for consideration.

The following corrective action options to assure continued safe storage of Vogtle spent fuel would be considered by GPC if unexpected degradation problems were detected:

1. The degraded Boraflex could be evaluated to determine whether the degradation and any expected future degradation would adversely affect GPC's ability to satisfy the 0.95 k-effective limit for the Vogtle spent fuel pool. If the pool could still satisfy this limit, no further action would be necessary.

2. Administrative controls on the enrichment and/or burnup of fuel to be placed in or adjacent to storage cell locations that have degraded Boraflex, or loading techniques such as checkerboard patterns, could be used to assure that the k-effective would remain less than or equal to the 0.95 limit.

3. A poison material such as a control rod or burnable poison could be added to any new fuel assembly to be placed in a storage cell with degraded Boraflex. This would reduce the k-effective to less than or equal to the 0.95 limit.

4. GPC has taken no credit for the soluble boron concentration in the spent fuel pool water. This boron concentration is capable of being maintained such that the k-effective is less than 0.95 with degraded Boraflex.

5. The storage cells with the degraded Boraflex could be blocked off to prevent loading of any fuel assembly into the cell.

via the transfer canal when the gate between the pool and canal is open.

B. Spent Fuel Pool Dewatering

The most serious failure of this system would be complete loss of water in the storage pool. In accordance with Regulatory Guide 1.13, the design of the SFPCPS limits the loss of coolant that could be caused by maloperation or failure of system components such that spent fuel does not become uncovered.

The spent fuel pool cooling pump suction connections are located near the normal water level so that the pool cannot be gravity drained. Each return line contains an antisiphon hole to prevent the possibility of gravity draining of the pool via these lines. Finally, the lines to and from the skimmer/strainers are located near the normal water level.

The accidental opening of the gate between the spent fuel pool and the transfer canal, if the canal is dry, would lower the water level approximately 6 ft, leaving about 18 ft of water over the top of the spent fuel assemblies.

Makeup water sources are provided to replace evaporative and minor leakage losses. These sources include the refueling water storage tank, the reactor makeup water storage tank, the demineralized water storage tank, and the recycle holdup tanks. Makeup to the spent fuel pit should be started upon a low-level alarm signal from the spent fuel pool level instrumentation. 7

From 9.1.3.5

The spent fuel pool, transfer canal, and spent fuel cask loading pit have stainless steel liners welded to embedments in the walls and floors. At every liner weld seam continuous drains are provided for leak detection. These are interconnected and drain to a collection point which is monitored to determine whether leakage is occurring.

C. Water Quality

Only a very small amount of water is interchanged between the refueling canal and the spent fuel pool, as fuel assemblies are transferred in the refueling process. Whenever a fuel assembly with defective cladding is transferred from the fuel transfer canal

FROM 9.1.3.5

to the spent fuel pool, a small quantity of fission products may enter the spent fuel cooling water. The purification loop removes fission products and other contaminants from the water. By maintaining radioactivity concentrations, excluding tritium, in the spent fuel pool water at or below 5×10^{-3} µCi/g for dominant gamma-emitting isotopes, the dose rate at the surface of the pool is 2.5 mrem/h or less.

9.1.3.6 Tests and Inspections

Active components of the SFPCPS are in either continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice.

No special equipment tests are required, since system components are normally in operation when spent fuel is stored in the fuel pool.

Sampling of the fuel pool water for gross activity and particulate matter concentration is conducted periodically. The layout of the components of the SFPCPS is such that periodic testing and inservice inspection of this system are possible. Details of the inservice inspection program are outlined in section 6.6.

A. Instrumentation Application

The instrumentation provided for the SFPCPS is discussed in the following paragraphs. Alarms and indications are provided as noted.

B. Temperature

Instrumentation is provided to measure the temperature of the water in the spent fuel pool and to give local indication as well as annunciation in the control room when normal temperatures are exceeded.

Instrumentation is also provided to give local indication of the temperature of the spent fuel pool water as it leaves either heat exchanger.

C. Pressure

Instrumentation is provided to measure and give local indication of the pressures in the spent fuel pool pump suction and discharge lines and in the skimmer pump discharge line. Instrumentation is also provided

For Safety Class 3 fuel handling and storage equipment, consideration is given to the OBE only insofar as failure of the Safety Class 3 equipment might adversely affect other safety-related equipment.

For nonnuclear safety equipment, design for the SSE is considered if failure might adversely affect safety-related equipment. Design for the OBE is considered if failure of the nonnuclear safety component might adversely affect safety-related equipment.

9.1.4.3.3 Containment Pressure Boundary Integrity

The fuel transfer tube which connects the refueling canal (inside the reactor containment) and the fuel storage area (outside the containment) is closed on the refueling canal side by a blind flange at all times except during refueling operations. Two seals are located around the periphery of the blind flange with leak-check provisions between them.

9.1.4.3.4 Radiation Shielding

During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 2.5 mrem/h or less. This is accomplished by maintaining a minimum of 10 ft of water above the top of the active fuel height during all handling operations.

The two fuel handling devices used to lift spent fuel assemblies are the refueling machine and the fuel handling machine. The refueling machine contains positive stops which prevent the fuel assembly from being raised above a safe shielding height. The hoist on the fuel handling machine and the containment fuel storage area crane moves spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to within this safe shielding height.

9.1.4.4 Inspection and Testing Requirements

The test and inspection requirements for the equipment in the LLHS are as follows:

A. Fuel Handling Machine, Refueling Machine and New Fuel Elevator

The minimum acceptable tests at the shop include the following:

TABLE 12.2.2-1 (SHEET 1 OF 3)

PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVE CONCENTRATIONS

Leak Rates (lb/day)	
Equivalent reactor coolant leak into containment during power for noble gases	5100
Equivalent reactor coolant leak into containment for halogens	5.1
Equivalent reactor coolant leak into auxiliary building	160
Equivalent reactor coolant leak into letdown heat exchanger valve gallery	7.4
Equivalent steam generator steam leak into turbine building	40,800

Evaporation Rates (g/min)	
From refueling pool into containment atmosphere	3240
From spent fuel pool into fuel building atmosphere	3920

Partition Factors	Noble Gases	Halogens	Particulates	Tritium
Auxiliary building	1	0.0075	0.0001	0.1
Fuel handling building	Negligible	Negligible	Negligible	1
Radwaste building	(a)	0.0075	0.0001	(a)

TABLE 12.2.2-1 (SHEET 2 OF 3)

Ventilation Rates (ft^3/min)	
Containment during power	5000
Containment during refueling	15,000
Fuel handling building during refueling	30,000
Auxiliary building	72,000
Auxiliary building letdown heat exchanger valve gallery	90
Turbine building	1.1×10^6
Radwaste solidification building	28,800
Volumes of the Regions (ft^3)	
Containment	2.75×10^6
Fuel handling building	5.0×10^5
Auxiliary building	1.9×10^6
Auxiliary building letdown heat exchanger valve gallery	2730
Turbine building	5.3×10^6
Radwaste solidification building	7.2×10^5
Radwaste transfer building	9×10^4

TABLE 12.2.2-1 (SHEET 3 OF 3)

Miscellaneous Information	
Failed fuel percentage for fission products	0.12
Reactor coolant specific activities	Table 11.1-7
Steam generator steam activities	Table 11.1-7
Plant capacity factor (percent)	80

a. The contribution to the airborne radioactivity concentration from noble gases and tritium in the radwaste buildings is considered negligible.

TABLE 12.2.1-19 (SHEET 2 OF 2)

Nuclide	Concentration for Maximum Failed Fuel (μCi/g)	Concentration for Expected Failed Fuel (μCi/g)
Ru-106	3.5438-09	2.5288-10
Te-125m	6.6049-09	6.8091-10
Te-127m	7.1135-08	6.8825-09
Te-129m	4.2167-07	3.1090-08
Te-131m	1.3749-08	1.3754-09
Te-132	1.6719-06	1.6204-[illegible]
Ba-140	7.3745-08	3.8577-09
La-140	1.9926-09	2.2546-10
Ce-141	1.4593-06	1.4593-06
Ce-143	4.2130-10	3.2189-11
Ce-144	7.5669-07	7.5669-07
Pr-143	1.1310-08	9.0264-10
Np239	--	3.9583-09
Ag-110m	3.5226-08	--

a. These activities are used to verify shield wall thicknesses. For dose assessment (section 12.4), activities in the pool are assumed to be limited administratively so that pool surface dose rates are less than 2.5 mrem/h.

TABLE 12.2.2-2 (SHEET 1 OF 3)

AIRBORNE RADIOACTIVITY CONCENTRATIONS
(μCi/cm³)

Nuclide	Containment (100% Power)	Containment (Refueling)	Fuel Handling Building (Refueling)	Turbine Building
H-3	1.50E-6	2.50E-6	2.50E-6	4.54E-10
N-16	-			-
Ar-41	9.97E-7			-
Mn-54	2.95E-10			-
Fe-59	1.01E-9-10			-
Co-58	1.01E-9			-
Co-60	4.56E-10			-
Br-83	1.67E-11			5.24E-16
Br-84	2.62E-12			7.74E-17
Br-85	2.98E-14			3.50E-19
Kr-83m	5.33E-8			2.46E-15
Kr-85m	4.46E-7			1.20E-14
Kr-85	5.40E-8			5.90E-16
Kr-87	1.16E-7			6.76E-15
Kr-88	6.60E-7			2.24E-14
Kr-89	5.25E-10			2.64E-16
Rb-86	-			3.36E-18
Rb-88	-			2.88E-16
Sr-89	2.28E-11			-
Sr-90	3.99E-12			-
I-130	1.70E-11			5.96E-16
I-131	3.05E-9			9.98E-14
I-132	3.26E-10			3.96E-14
I-133	3.45E-9			1.09E-13
I-134	7.22E-11			2.26E-15
I-135	1.20E-9			3.72E-14
Xe-131m	1.65E-7			1.86E-15
Xe-133m	8.63E-7			1.09E-14
Xe-133	4.39E-5			4.98E-14
Xe-135m	6.06E-9			1.23E-15
Xe-135	1.85E-6			3.42E-14
Xe-137	1.12E-9			5.22E-16
Xe-138	1.98E-8			4.24E-15
Cs-134	2.95E-10			7.72E-16
Cs-136	-			4.32E-16
Cs-137	5.10E-10			6.36E-16
Ba-137m	-			1.89E-11

TABLE 12.2.2-2 (SHEET 2 OF 3)

	Auxiliary Building		Radwaste Transfer Building	
Nuclide	Corridor	Letdown Heat Exchanger Valve Gallery	Corridor	Spent Resin Tank
H-3	2.48E-9	9.19E-8	-	-
N-16	-	-	-	-
Mn-54	-	-	9.31E-15	1.11E-13
Fe-59	-	-	2.18E-15	2.65E-14
Co-58	-	-	1.13E-13	1.36E-12
Co-60	-	-	5.14E-14	6.17E-13
Br-83	8.93E-13	3.25E-11	-	-
Br-84	3.59E-13	1.25E-11	-	-
Br-85	9.13E-15	2.92E-13	-	-
Kr-83m	4.49E-10	1.62E-8	-	-
Kr-85m	2.23E-9	8.18E-8	-	-
Kr-85	1.19E-10	4.41E-9	-	-
Kr-87	1.24E-9	4.47E-8	-	-
Kr-88	4.26E-9	1.55E-7	-	-
Kr-89	2.13E-11	6.83E-10	-	-
Rb-86	2.20E-16	8.17E-15	4.91E-15	5.55E-14
Rb-88	2.87E-13	9.76E-12	-	-
Sr-89	-	-	3.08E-14	3.74E-13
Sr-90	-	-	4.09E-15	4.93E-14
I-130	4.17E-13	1.54E-11	-	-
I-131	5.19E-11	1.93E-9	2.80E-11	3.41E-10
I-132	1.81E-11	6.58E-10	-	-
I-133	7.32E-11	2.71E-9	-	-
I-134	7.36E-12	2.61E-10	-	-
I-135	3.73E-11	1.37E-9	-	-
Xe-131m	3.71E-10	1.38E-8	-	-
Xe-133m	2.12E-9	7.85E-8	-	-
Xe-133	1.01E-7	3.76E-6	-	-
Xe-135m	1.61E-10	5.42E-9	-	-
Xe-135	6.73E-9	2.48E-7	-	-
Xe-137	4.42E-11	1.42E-9	-	-
Xe-138	5.40E-10	1.82E-8	-	-
Cs-134	6.44E-14	2.39E-12	6.46E-12	6.18E-11
Cs-136	3.47E-14	1.29E-12	1.34E-13	1.38E-12
Cs-137	4.71E-14	1.75E-12	5.33E-12	5.17E-11
Ba-137m	1.09E-16	1.82E-13	3.97E-12	4.90E-11

TABLE 12.2.2-2 (SHEET 3 OF 3)

	Radwaste Solidification Building	
Nuclide	Corridor	Spent Resin Tank
H-3	-	-
N-16	-	-
Mn-54	1.92E-15	2.25E-13
Fe-59	4.52E-16	5.31E-14
Co-58	2.33E-14	2.73E-12
Co-60	1.06E-14	1.24E-12
Br-83	-	-
Br-84	-	-
Br-85	-	-
Kr-83m	-	-
Kr-85m	-	-
Kr-85	-	-
Kr-87	-	-
Kr-88	-	-
Kr-89	-	-
Rb-86	1.01E-15	1.11E-13
Rb-88	-	-
Sr-89	6.34E-15	7.49E-13
Sr-90	8.43E-16	9.89E-14
I-130	-	-
I-131	5.77E-12	6.83E-10
I-132	-	-
I-133	-	-
I-134	-	-
I-135	-	-
Xe-131m	-	-
Xe-133m	-	-
Xe-133	-	-
Xe-135m	-	-
Xe-135	-	-
Xe-137	-	-
Xe-138	-	-
Cs-134	1.33E-12	1.24E-10
Cs-136	2.77E-14	2.77E-12
Cs-137	1.10E-12	1.04E-10
Ba-137m	8.20E-13	9.85E-11

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard area is regulated and controlled by radiation zoning and access control (section 12.5). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. During plant operation, personnel normally gain access to radiation controlled areas through the access control building.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. The radiation zone categories employed and their descriptions are given in table 12.3.1-1. The zoning for each plant area under normal conditions is shown in figure 12.3.1-1. The zoning for each plant under accident conditions is shown in figure 12.3.1-2. Radiation zones shown in the figures are based upon conservative design data. Actual in-plant zones and control of personnel access will be based upon surveys conducted by health physics as described in section 12.5.

In accordance with Section II.B.2 of NUREG-0737, a radiation and shielding design review was performed to identify vital areas and equipment. Areas which may require occupancy to permit an operator to aid in the long term recovery from an accident are considered as vital. Vital areas include the control room, technical support center, safety-related motor control centers and switchgear in the control building, auxiliary building, diesel generator building, auxiliary feedwater pumphouse, radiochemistry laboratory, and the remote shutdown panels. Projected dose rates for these vital areas at various times after an accident are given in table 12.3.1-5. 4
VEGP is designed to ensure the capability to achieve cold shutdown without subjecting personnel to excessive radiation exposure. This capability is further described in section 7.4. Radiation protection design features and access controls are described in sections 12.3 and 12.5. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates defined in figure 12.3.1-2 and table 12.3.1-5, and appropriate time limits 4
for presence in the area are imposed.

Ingress or egress of plant operating personnel to controlled access areas is controlled by the plant health physics staff to ensure that radiation levels and exposures are within the limits prescribed in 10 CFR 20. Any area having a radiation level that could cause a whole body exposure in any 1 h in excess of 5 mrem, or in any 5 consecutive days in excess of 100 mrem, will be posted "Caution, Radiation Area." Radiation areas are provided with access alert barriers, e.g., chain, rope, door, etc. Any area having a radiation level that could cause whole body exposure in any 1 h in excess of 100 mrem will be posted "Caution, High Radiation Area." High radiation areas (> 1000 mrem/h) are provided with locked or alarmed barriers. For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/h at 45 cm that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. During periods when access to a high radiation area is required, positive control is exercised over each individual entry. To the extent practicable, the measured radiation level and the location of the source is posted at the entry to any radiation area or high radiation area. 29

Posting of radiation signs, control of personnel access, and use of alarms and locks are in compliance with requirements of 10 CFR 20.203. The flow of personnel is shown in figure 12.3.1-3. 29

Each access door to a high radiation area is equipped with a single automatically controlled access terminal (ACAT). Access into these high radiation areas is accomplished by inserting a card device into the computer control ACAT. Entry into a high radiation area is displayed at the health physics console in the health physics station.

assuming that stringent water chemistry control and improved design will minimize crud buildup and hence the expected dose rates in various radiation zones, and by the recognition that real maximum doses in a given zone are localized effects. The expected average doses given above are used in computing the doses for personnel involved in all operations, except inservice inspection and special maintenance.

The direct radiation dose estimates have been developed from exposure models for each of the major job categories within routine functions. Each exposure model has been developed by breaking the job into individual packages and identifying expected radiation fields, time spent in each radiation field, and the number of men required to carry out each package. Engineering judgment and feedback from operating plant experience have been used to define typical values for each parameter in the exposure model. As such, the resultant exposure estimates should be used as typical values, keeping in mind the variability of the input data from which the estimates were developed.

Exposure to plant personnel from direct gamma radiation during the performance of routine functions is estimated to be approximately 418 man-rem/year/unit. Details of the man-rem estimates are given in table 12.4.1-13.

12.4.1.2 Airborne Radioactivity Dose Estimates

Due to leakages of radioactive fluids into the auxiliary, containment, radwaste, fuel handling, and turbine buildings, plant personnel are exposed to radionuclides released into the atmosphere of these buildings by the leaked fluids. These atmospheric contaminants contribute to the total body, thyroid, and lung doses.

The peak airborne concentrations for most areas in the plant are within the limits specified in 10 CFR 20. By use of appropriate respiratory equipment and/or limitation of occupancy time, personnel are allowed to enter areas where the airborne activity levels exceed 10 CFR 20 limits.

The expected annual doses to plant personnel from airborne radioactivity for each building in the plant are presented in table 12.4.1-14. The assumptions used to determine airborne radioactivity in each building, along with the airborne concentrations for all areas, are presented in subsection 12.2.2 and tables 12.2.2-1 and 12.2.2-2.

Doses resulting from airborne radioactivity are calculated by the methods discussed below using appropriate portions of

TABLE 12.4.1-14

DOSES TO PLANT PERSONNEL CAUSED BY AIRBORNE RADIOACTIVITY

Location	Assumed Occupancy (h/year)	Total Body Gamma Dose (man-rem/year)	Lung Dose (man-rem/year)	Inhalation Thyroid Dose (man-rem/year)	Airborne Tritium Dose (man-rem/year)
Auxiliary building corridor	2000 (40 h/week-50 weeks/year)	2.68E-2	3.93E-3	2.80E-1	2.55E-3
Auxiliary building letdown heat exchanger valve gallery	50	2.46E-2	3.64E-3	2.60E-1	2.37E-3
Turbine building	2000 (40 h/week-50 weeks/year)	5.52	6.90-6	4.96-4	4.68-4
Fuel building H-3 only	168 (56 h/week-3 weeks/year)	NA	NA	NA	2.16E-1
Containment (full power)	46	3.59E-1	9.12E-3	6.55E-1	6.67E-2
Containment (refuel) H-3 only	168 (56 h/week-3 weeks/year)	NA	NA	NA	2.16E-1
Radwaste solidification building	2000 (40 h/week-50 weeks/year)	3.58E-6	2.31E-3	2.24E-2	NA
Radwaste transfer building	50 (1 h/week-50 weeks/year)	4.34E-7	2.80E-4	2.72E-3	NA

3