GE Nuclear Energy

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April 16, 1993

Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **DFSER Chapter 12 Outstanding Items**

Dear Chet:

Enclosed are SSAR markups addressing Confirmatory Item 12.2.3-1 and COL Action Items 12.1.1-1, 12.1.3-1, 12.3.4-1, 12.3.4-2, 12.3.4-3, 12.5-1 and 12.5.1-1.

Please provide a copy of this transmittal to Roger Pedersen.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Hal Careway (GE)
Norman Fletcher (DOE)
Terry Hosler (GE)

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CHAPTER 12

TABLE OF CONTENTS

SECTION 12.1

CONTENTS (Continued)

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

12.1.1 Policy Considerations

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel will be kept as low as reasonably achievable (ALARA).

12.1.1.1 Design and Construction Policies

The ALARA philosophy was applied during the initial design of the plant and implemented via internal design reviews. The design was reviewed in detail for ALARA considerations and was reviewed, updated and modified as necessary during the design phase as experience was gained from operating plants. Engineers reviewed the plant design and integrated the layout, shielding, ventilation and monitoring instrument designs with traffic control, security, access control and health physics aspects to ensure that the overall design is conducive to maintaining exposures ALARA.

All pipe routing containing radioactive fluids were reviewed as part of the engineering design effort. This ensured that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure to personnel.

Operating plant results were continuously integrated during the design phase of the ABWR Standard Plant.

12.1.1.2 Operation Policies

Out of ABWR Standard Plant scope.

12.1.1.3 Compliance with 10CFR20 and Regulatory Guides 8.8, 8.10 and 1.8

Compliance of the ABWR design with Title 10 of the Code of Federal Regulations, Part 20 (10CFR20), is ensured by the compliance of the design and operation of the facility within the guidelines of Regulatory Guides 8.8, 8.10, and 1.8.

12.1.1.3.1 Compliance with Regulatory Guide 8.8

The policy considerations regarding plant operations contained in Regulatory Guide 8.8 are out of ABWR Standard Plant scope. See Subsection 12.1.4.4 for COL license information.

The design of the ABWR Standard Plant fully meets the intent of Regulatory Guide 8.8, and reflects the commitment of General Electric. Examples of compliance with Section C.2 of Regulatory Guide 8.8 are delineated in Subsection 12.3.1. Design features of the ABWR allow easy compliance with the recommendations of Subsections C.2 and C.4 of the guide. For instance, provisions are made in systems such as the reactor water cleanup system (RWCS) to allow flushing of the piping in shielded cubicles before entry, and to use remote reach rods. Portable breathing air is utilized in those areas where past experience indicates airborne radioactivity has been a problem. Design provisions allow for remote operation of fuel handling and radwaste cask filling.

12.1.1.3.2 Compliance with Regulatory Guide 8.10

Out of ABWR Standard Plant scope. See Subsection 12.1.4.1 for COL license information.

12.1.1.3.3 Compliance with Regulatory Guide 1.8

Out of ABWR Standard Plant scope. See Subsection 12.1.4.2 for COL license information.

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA are presented in detail in Subsections 12.3.1 and 12.3.2.

12.1.2.1 General Design Consideration for ALARA Exposures

General Design considerations and method employed to maintain inplant radiation exposures ALARA, consistent with the recommendations of Regulatory Guide 8.8, have two objectives:

(1) minimizing the necessity for and amount of personnel time spent in radiation areas, and

(9) providing space for pumps and valves outside of highly radioactive areas;

(10) providing remotely operated centrifugal discharge and/or backflushable filter systems for highly radioactive radwaste and cleanup systems;

(11) providing labyrinth entrances to radioactive pump, equipment, and valve rooms;

(12) providing adequate space in labyrinth entrances for easy access;

(13) maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity; and

(14) providing both automatic logic control and mechanical stop devices for control of the transverse in-core (TIP) probe to prevent withdrawl of the radioactive portions of the TIP onto the cable spoolers.

12.1.3 Operational Considerations

Out of ABWR Standard Plant scope. See Subsection 12.1.4.3 for COL license information.

12.1.4 COL License Information

12.1.4.1 Regulatory Guide 8.10

Compliance with Regulatory Guide 8.10 shall be demonstrated by the COL applicant (See Subsection 12.1.1.3.2).

12.1.4.2 Regulatory Guide 1.8

Compliance with Regulatory Guide 1.8 shall be demonstrated by the COL applicant (See Subsection 12.1.1.3.3).

12.1.4.3 Occupational Radiation Exposures

to the level of detail provided in Regulatory Guide 1.70,

COL applicants will provide the criteria and/or conditions under which various operating procedures and techniques shall be provided to ensure that occupational radiation exposures one ALARA are implemented (See Subsection 12.1.3).

12.1.4.4 Regulatory Guide 8.8

Compliance with Regulatory Guide 8.8 shall be demonstrated by the COL applicant. (See Subsection 12.1.1.3.1)

CI 12.2.3-1

With respect to the reactor building, the overall plant design has divided the reactor building into three separate and independent divisions. ECCS components are contained in each division in separate isolated rooms such that the failure of one system in one division will not affect in any way components in another division. Releases of radioactive material either in the form of water or steam (airborne) are contained in and isolated to a large extent in the compartment in which it might occur by the use of water tight doors and area radiation monitors which isolate the HVAC system from the compartment. Divisional separation under such conditions is complete. Sumps are designed to detect and alarm in the event of leaks in excess of one gallon per minute establishing a threshold for leak before break on the larger water carrying piping systems. All connections to the primary containment not terminating in the reactor building meet GDC54, 55, 56, and 57. Therefore, in the event of an accident involving radioactive sources in the primary containment or reactor building such sources would be contained and isolated for further treatment and decontamination.

Likewise potential releases in the radwaste building will be contained by isolating the radwaste building atmosphere and sealing any water releases in the building which is seismically qualified and steel lined to prevent any potential water releases. Such potential releases are discussed in Section 15.7.

The turbine building contains no major sources of releasable radioactivity (discounting N-16 because of the 7.7 second half-life) and potential releases are limited to liquid releases of low activity water from the feedwater and condenser system. Two other sources exist which contain radioactivity species but in form not amenable for release. The potential for accident sources from these two sources, the offgas system and condenser demineralizers, is reduced due to heavy shielding and compartmentalizing these components.

Estimates on sources and location for limiting design basis events are found in Chapter 15 and sources for degraded core events as a function of probability are found in Chapter 19.

12.2.1.3 Turbine Building Sources

left for construction specific details as provided in DAC Table 3.7.

Turbine building sources are primarily dominated by N-16 in the steam flow from the pressure vessel. The N-16 source results in significant gamma shine from the main steam lines and steam bearing components (turbines, moisture separators, and reheaters) on the order of 20-50 rad/hr contact. Estimates of typical BWR sources and gamma shine are given in Reference 11. Since the geometry of the radiation source is dependent on the exact turbine configuration used, the specific details for the turbines and turbine reheaters are ~~interface requirements for referencing applicant as called out in Subsection 12.2.4~~. Tables 12.2-26 through 12.2.-28 provide estimates of inventories for the moisture separator, condenser, and condenser demineralizer. The offgas system is divided into three major components, steam jet air ejector (SJAE), recombiner, and charcoal tanks. The inventory in the SJAE is given in Table 12.2-29 while the inventories in the recombiner and charcoal tanks are given in Table 12.2-14. The offgas system is more fully described in Subsection 12.2.1.2.6.3.

12.2.2 Airborne and Liquid Sources for Environmental Consideration

This subsection deals with the source and parameters required to evaluate airborne and liquid releases during normal plant operations for compliance with 10CFR20 and 40CFR190. In addition, specific sources are addressed with regard to airborne contamination in the refueling area under Subsection 12.2.2.3 for evaluation of worker potential doses under 10CFR20. However, for compliance to worker airborne limitations as stipulated in 10CFR20, direct evaluations are ~~not contained in this document, but are~~ left as COL license ~~information (see Subsection 12.2.3.1) as stipulated in ITAAC Table 3.7b.~~ Likewise, the complete evaluation for compliance to GDC61 for the fuel handling area is also ~~COL license information based upon final fuel configuration and in compliance with ITAAC~~ Table 3.7a.

required in DAC Table 3.7.

SECTION 12.2

CONTENTS (Continued)

12.2.2.3 Airborne Sources During Refueling

The airborne radioactivity during refueling in the containment is expected to be similar to that observed in operating stations. Experience at operating BWR has shown that airborne radioactivity can result from the water in the reactor cavity exceeding 100 F and flaking of cobalt dioxide (CoO_2) from the 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 and internals removal have been determined from operating plant experience. The major radioisotopes found were I-131, Co-60, and Mn-54, with Nb-95, Zr-95, Ru-103, and Ce-144 at moderate concentrations, and with Ce-141, Cs-137, Co-58, and Cr-51 at low concentrations. The radioactive particulates ranged as high as 2×10^{-8} μCi/cc and the I-131 as high as 4×10^{-8} μCi/cc.

To minimize the containment airborne radioactivity contribution due to removal of the reactor pressure vessel head:

(1) the steam dryer and separator surfaces will be kept wet or covered;

(2) the fuel pools are cooled through heat exchangers of large capacity; and the

(3) ventilation system on the refueling pool is designed to sweep air from the pool surface and remove a large portion of potential airborne contamination.

12.2.2.4 Average Annual Doses

For compliance with 10CFR50, Appendix I, evaluations have been made to determine average annual doses to unrestricted areas subject to airborne and liquid releases. For airborne dose calculations, isotopic release were taken from Table 12.2-20 assuming a 0.5 mile exclusion boundary. Releases were assumed from plant stack since all major (reactor building, turbine building and radwaste building) ventilation systems pipe to the stack for normal releases. Since a site meteorology is not definitively defined, astatistical approach was used to evaluate the releases over a series of metrologies discussed in References 6 and 7. Doses were calculated using methodologies and conversion factors consistent with Regulatory Guides 1.109 and 1.111 as implemented in References 8 and 9. The results of the airborne evaluations is given in Table 12.2-21. For the ingestion doses given in Table 12.2-21, ingestion values given in Table E-5 of Regulatory Guide 1.109 were used.

required in DAC Table 3.7

The evaluations above provide airborne sources and offsite doses for compliance with 10CFR50, Appendix I. For complete evaluations for compliance to 40CFR190, gamma shine evaluations are not contained in this document since adequate detail for skyshine evaluations from the turbine complex are COL applicant supplied. Therefore, skyshine evaluations are left as COL license information under Subsection 12.2.3.2 as stipulated in ITAAC 3.7a.

12.2.2.5 Liquid Releases

The ABWR is designed not to release radioactive liquid effluents. However, under certain conditions of high water inventory, up to 0.1 Curie per year excluding tritium as described in Subsection 11.2.3. These releases are given in Table 12.2-22 and form the basis for estimating doses using methodologies consistent with Regulatory Guide 1.113 as implemented in Reference 10. The results of the liquid release assuming dilution factors described in Subsection 11.2.3.2 are shown in the dose evaluation in Table 12.2-23.

12.2.3 COL License Information

12.2.3.1 10CFR20 and GDC61 Compliance

COL applicants will provide source tables and operational criteria to insure compliance with respect to worker restrictions of 10CFR20 and GDC61. (See Subsection 12.2.2)

12.2.3.2 Turbine Building Compliance

COL applicants will provide gamma shine calculations for the turbine complex to insure

turbine building gamma shine both offsite and to surrounding buildings is within applicable limits. (See Subsection 12.2.2.4)

12.2.~~4~~ 3 References

1. J.E. Smith, *Noble Gas Experience in Boiling Water Reactors*, Paper No. A-54, presented at Noble Gases Symposium, Las Vegas, Nevada, September 24, 1974.

2. *Airborne Releases from BWRs for Environmental Impact Evaluations*, NEDO-21159-2 (1977).

3. American Nuclear Society, ANS-18.1, Table 5.

4. *Airborne Releases from BWRs for Environmental Impact Evaluations*, NEDO-21159 March 1976.

5. *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors* (BWR-GALE Code) U.S. NRC NUREG-0016 Rev. 1, January 1979.

6. I. Hall, et al, *Generation of Typical Meteorological Years for 26 SOLMET Stations,* Sandia National Laboratory Report SAND78-1601 (1978).

7. D.C. Aldrich, et al, *Technical Guidance for Siting Criteria Development,* NUREG/CR-2239 (1981).

8. E.W. Bradley, *Gamma and Beta Dose to Man from Noble Gas Release to the Atmosphere GEMAN Code.* NEDO-25132A, April, 1980.

9. E.W. Bradley and V.D. Nguyen, *Radiation Exposure from Airborne Effluents - the REFAE code,* NEDO-25257, July, 1980.

10. P.P. Standcavage and D.G. Abbott, *Liquid Discharge Doses LIDSR Code,* NEDM-20609-01, Aug, 1976.

11. Rogers, D.R., *BWR Turbine Equipment N-16 Radiation Shielding Studies*, GE NEDO-20206, December 1973.

SECTION 12.3

CONTENTS (Continued)

TABLES

12.3.7.1 Airborne Radionuclide Concentration Calculations
12.3.7.2 Operational Considerations
12.3.7.3 Requirements of 10CFR70.24

of concrete or its equivalent (other material or distance) is required on any ray pathway from the main steam lines to any point which may be inhabited during normal operations. The design of the steam tunnel is shown on Figures 1.2-14, 1.2-15, 1.2-20, 1.2-21, and 1.2-28. The tunnel is classified as Seismic Category I in the reactor building and in the control building and is designed to UBC Seismic Standards in the turbine building. The interface between the buildings provides for bayonet connection to permit differential building motion during seismic events and shielding in the areas between buildings. The exact details on the bayonet design are not shown on the referenced arrangement drawings but requires complete shielding in the building interface area. The tunnel also serves a secondary purpose as a relief and release pathway for high energy events in the reactor building. Any high energy event (line break) in the reactor building will, through a series of blow out panels, vent into the steam tunnel and from the steam tunnel through the tunnel vent shaft to the turbine building (see Figure 1.2-28) for processing to the plant stack. See Subsection 6.2.3.3.1 for more complete description of this function.

12.3.3 Ventilation

The HVAC systems for the various buildings in the plant are discussed in Section 9.4, including the design bases, system descriptions, and evaluations with regard to the heating, cooling, and ventilating capabilities of the systems. This section discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

(1) The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.

(2) The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance shall be kept below the limits of 10CFR20 during normal power operation. This is accomplished by establishing in each area a reasonable compromise between specifications on potential airborne leakages in the area and HVAC flow through the area. Appendix 12A to this chapter outlines the methodology by which such calculations are made. The COL applicant is responsible for performing calculations for the expected airborne radionuclide concentrations to verify the adequacy of the ventilation system. See Subsection 12.3.7.1 for COL license information.

The applicable guidance provided in Regulatory Guide 1.52 has been implemented for the ESF filter systems for the control building outdoor air cleanup system and the standby gas treatment system (STGS) as described in Subsections 6.5.1 and 9.4.1.

12.3.3.2 Design Description

In the following sections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. wg) at all times, except if exhausting or isolation are required, in order to prevent infiltration of contaminants. Fresh air is taken in via a dual inlet system, which has both intake structures on the roof of the building. The inlets are arranged with respect to the SGTS exhaust stack such that at least one of the intakes is free of contamination after a LOCA. Both inlets, however, can be submerged in contaminated air from a LOCA, but the calculated dose in the control room from such an eventuality is still below the limit of Criterion 19 of 10CFR50, Appendix A.

Outside air coming into the intakes is normally filtered by a particulate filter. If a high radiation level in the air is detected by the airborne radiation monitoring system, flow is automatically diverted to another filter train (an outdoor air cleanup unit) that has:

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The following systems are provided to monitor area radiation and airborne radioactivity within the plant:

(a) The area radiation monitoring system (D21/ARM) continuously measure, indicate and record the gamma radiation levels at strategic locations throughout the plant except within the primary containment, and activate alarms locally as well as in the control room on high levels to warn operating personnel to avoid unnecessary or inadvertent exposure. This system is classified as non-essential.

(b) The containment atmospheric monitoring system (D23/CAM) continuously measure, indicate, and record the gamma radiation levels within the primary containment (drywell and suppression chamber), and activate alarms in the main control room on high radiation levels. As described in Section 7.6.2, four gamma sensitive ion chamber channels are provided to monitor gamma radioactivity in the primary containment during normal, abnormal and accident conditions. Each of the four monitoring channels cover the range from 1R/hr to 10^7 R/hr. The CAM system is classified as safety-related.

(c) The airborne radioactivity in effluent releases and ventilation exhausts is continuously sampled and monitored by the process radiation monitoring system (D11/PRM) for noble gases, air particulates and halogens. As described in Section 11.5, the presence of airborne contamination is sampled and monitored at the stack common discharge, in offgas releases, and in the ventilation exhaust from buildings. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used for compliance with 10CFR20 restrictions. This portable system is designed to meet the criteria of DAC 3.7b and monitors airborne radioactivity in work areas prior to entry where potential levels exist that may exceed the allowable concentration limits. The instrumentation provided to monitor airborne radioactivity is classified as non-essential, and is the responsibility of the COL applicant. See Subsection 12.3.7.2 for COL license information.

12.3.4.1 ARM System Description

The area radiation monitoring system consists of gamma sensitive detectors, digital area radiation monitors, local auxiliary units with indicators and local audible warning alarms, and recording devices. The detector signals are digitized and optically multiplexed for transmission to the radiation monitors in the main control room. Each ARM radiation channel has two independently adjustable trip alarm circuits, one is set to trip on high radiation and the other is set to trip on downscale indication (loss of sensor input). Also, each ARM monitor is equipped with self test feature that monitors for gross failures and will activate an alarm on loss of power or when a failure is detected. Auxiliary units with local alarms are provided in selected local areas for radiation indication and for activating the local audible alarms on abnormal levels. Each area radiation channel is powered from the non-1E vital 120 Vac source which is continuously available during loss of off-site power. The recording devices are powered from the 120 Vac instrument bus.

12.3.4.2 ARM Detector Location and Sensitivity

The location of each area detector is shown on the plant layout drawings for each building, Figures 12.3-56 through 12.3-73. The specific area radiation channels for each building are listed in Tables 12.3-3 through 12.3-7, along with reference to map location of the detector, the channel sensitivity range, and the areas for the local alarms. The range and sensitivity of each area radiation channel is classified as follows:

a. Range 10^{-2} mR/hr to 10^2 mR/hr - H (High Sensitivity)

b. Range 10^{-1} mR/hr to 10^3 mR/hr - M (Medium Sensitivity)

c. Range 1 mR/hr to 10^4 mR/hr - L (Low Sensitivity)

d. Range 10^2 mR/hr to 10^6 mR/hr - LL (Low Low Sensitivity)

e. Range 10^{-1} R/hr to 10^4 R/hr - VL (Very Low Sensitivity

12.3.4.3 Pertinent Design Parameters and Requirements

Two high-range radiation channels are provided to monitor radiation from accidental fuel handling. One detector is positioned near the fuel pool and the other located in the fuel handling area. Criticality detection monitors are not needed to satisfy the criticality accident requirements of 10CFR70.24~~, because the~~ when ~~ABWR design utilizes~~ specialized high density fuel storage racks ~~that~~ preclude the possibility of criticality accident under normal and abnormal conditions. The new fuel bundles are stored in racks that are located in the fuel vault while the spent fuel bundles are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded fuel storage racks are designed to be subcritical as defined in Sections 9.1 and 9.2. The COL applicant must verify and certify that the design meets the criteria specified in subsection 12.3.7.3.

COL 12.3.4-2

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV to 7 MeV. The energy dependence will not exceed 20% of point from 100 keV to 3 MeV. The overall system design accuracy is within 9.5% of equivalent linear full scale recorder output for any decade.

COL 12.3.4-1

The alarm setpoints will be established in the field, by the COL applicant, as specified in subsection 12.3.7.2, following equipment installation at the site. The exact settings will be based on sensor location, back ground radiation levels, expected radiation levels, and low occupational radiation exposures. The high radiation alarm setpoint for each channel is set slightly above the background radiation level that is normal to the area.

The area radiation monitoring instrumentation is designed to provide early detection and warning for personnel protection to insure that occupational radiation exposures will be as low as is reasonably achieved (ALARA) in accordance with guidelines stipulated in R.G. 8.2 and R.G. 8.8.

The area radiation monitoring system includes instrumentation provided to assess the radiation conditions in crucial areas in the reactor building (the RHR equipment areas) where access may be required to service the safety related equipment during post LOCA per R.G. 1.97.

12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the control room, the technical support center, the remote shutdown panel, the primary containment sample station (post accident sample system), the health physics facility (counting room), and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations in Subsection 15.6.5 are within regulatory guidelines.

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

During a design basis accident event, access to remote shutdown panel, nitrogen bottles, and the PASS and monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained under filtered positive pressure so that personal protection equipment (radiation protection suits, breathing gear, etc.) will be required in the access corridor. Primary contamination would occur from leakage through the PASS system and air infiltration from the environment. Both pathways are considered minimal and minor contamination under even the most adverse conditions is expected.

The reactor building vital areas are all located off one of of the two primary access ways except the nitrogen bottle areas which are located on the refueling floor and are accessible from the clean access corridor at the 4800 level (B1F) and up three floors to the 23500 level (3F). There are two access corri- dors, clean and dirty, with contamination in those areas limited to air inflitration from the environment and penetration leakage from the PASS system. In addition, the lines penetrating the PASS room are doubly valved permitting line isolation in the event of any potential rupture. Sources of radiation therefore are limited to minor leakage and gamma shine including the stack monitor room which contains only instrumentation and associated penetrations for monitoring stack effluent.

12.3.6 Post-Accident Radiation Zone Maps

The post-accident radiation zone maps for the areas in the reactor building are presented in Figures 12.3-12 through 12.3-22. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. These dose rates do not include the airborne contribution in the reactor building.

Post-accident zone maps of the control building and turbine building are presented in Figures 12.3-54 and 55 respectively. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.2.10.

~~12.3.7 Deleted~~

← INSERT A

12.3.8 References

1. N. M. Schaeffer, *Reactor Shielding for Nuclear Engineers*, TID-25951, U.S. Atomic Energy Commission (1973).

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

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

4. *Reactor Handbook*, Volume III, Part B, E.P. Blizzard, U.S. Atomic Energy Commission (1962).

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12.3.7 COL License Information

12.3.7.2 Operational Considerations

COL 12.3.4-1

Area radiation monitoring operational considerations, such as monitor alarm setpoints, listed in R.G. 1.70 are the COL applicants responsibility. Airborne radiation monitoring operational consideration such as the procedures for operation and calibration of the monitors as well as the placement of the portable monitors are also the COL applicants responsibility. (See Subsection 12.3.4.)

12.3.7.3 Requirements of 10CFR70.24

COL 12.3.4-2

COL applicants will provide information showing that their plant meets the requirements of 10CFR70.24 or request an exemption from this 10CFR70.24 requirement (See Subsection 12.3.4.3).

12.3.7.1 Airborne Radionuclide Concentration Calculatio

COL 12.3.4-3

The COL applicants will provide the calculations of the expected concentrations of the airborne radionuclides for the requisitioned ABWR plant design. (See Subsection 12.3.3.1.)

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Operational Considerations

Out of ABWR Standard Plant Scope. See Subsection 12.5.3.1 for COL license information.

12.5.2 In-Plant and Airborne Radioactivity Monitoring

Out of ABWR Standard Plant Scope. See Subsection 12.5.3.2 for COL license information.

12.5.3 COL License Information

12.5.3.1 Radiation Protection Program

COL applicants will provide, to the level of detail ~~provided in~~ required by Regulatory Guide 1.70, the implementation of a radiation protection program for operational considerations. (See Subsection 12.5.1)

12.5.3.2 Compliance with Paragraph 50.34(f)(xxvii) of 10CFR Part 50 and NUREG-0737 Item III.D.3.3

COL applicants will provide the portable instruments in operating reactors that accurately measure radio-iodine concentrations in plant areas under accident conditions and will provide training and procedures on the use of these instruments in compliance with Paragraph 50.34(f)(xxvii) of 10CFR Part 50 and NUREG-0737 Item III.D.3.3. (See Subsection 12.5.2)