CHAPTER 12

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CHAPTER 12

RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

12.1.1 Policy Considerations

BVPS management is committed to maintaining occupational radiation exposure (i.e., Total Effective Dose Equivalent-TEDE) as low as reasonably achievable (ALARA). This includes maintaining the annual dose to individuals working at the station ALARA, and keeping the annual integrated dose to station personnel ALARA. It is an established policy to perform all reasonable actions necessary to reduce radiation exposure during work activities at the Beaver Valley Power Station (BVPS). This policy was implemented during the design and construction stage as well as during the operation and maintenance of the plant.

Many features have been incorporated into the Beaver Valley Power Station - Unit 2 (BVPS-2) design to minimize radiation exposure. These features minimize crud development and crud traps, provide separation and shielding between radioactive components, utilize remote handling equipment, allow draining and flushing of components, and prevent the spread of airborne contaminants. Section 12.3 discusses these design features in more detail.

The design and construction of the facility is such that occupational exposures will be ALARA. This will include ensuring, to the extent practicable, that:

1. Design concepts and station features reflect consideration of the personnel activities that can be anticipated which lead to exposure to substantial sources of radiation and that station design features have been provided to reduce the anticipated exposure of station personnel to these sources of radiation.

2. Specifications for equipment reflect the objectives of ALARA, including among others, consideration of reliability, serviceability, and limitation of internal accumulation of radioactive material.

During the operation of BVPS-2, a comprehensive Radiation Protection Program will be implemented utilizing the guidance presented in Regulatory Guides 1.8 (1975), 8.8 (1978), and 8.10 (1975), as well as current reports of the International Commission on Radiological Protection. Title 10 of the Code of

Federal Regulations, Part 20 provides the regulatory framework under which the ALARA philosophy is established.

A radiation protection manager is given responsibility for the development, coordination, maintenance, and effective implementation of the BVPS ALARA Program implemented via assignment of appropriate supervisory personnel to perform radiological control tasks as required. The manager is also responsible for controlling radiation exposure in a manner consistent with ALARA requirements and is specifically responsible for the on-site radiation protection program. Responsibilities with respect to the ALARA Program include: participating in the selection of specific goals and objectives for the station, maintaining and implementing the station ALARA Program, and expediting the collection and dissemination of data and information concerning the program to management, and ensuring the radiation protection procedures are utilized, maintained current, and provides clearly established control policies, procedures, guidelines, limits and other pertinent information.

12.1.2 Design Considerations

This section discusses the methods and features by which the ALARA policy considerations of Section 12.1.1 were applied during the design and construction of BVPS-2.

12.1.2.1 Previous Experience

Experience from past and present engineering designs was evaluated to determine if and how equipment or facility designs could be improved to reduce personnel radiation exposure. In the design and construction of BVPS-2 this philosophy was developed and supported as follows:

1. Through feedback from current projects. The architect engineer's organizational arrangement provided excellent communication between radiation engineers assigned to different projects working within the same staff group.

2. Through the use of problem reports generated by the architect/engineer or the results of known or reported design deficiencies in operational plants or current projects. These reports required a response from each current project describing the design change or administrative control to be incorporated to ensure ALARA.

3. Through the use of available sources of information such as U.S. Nuclear Regulatory Commission (USNRC) event reports and operating experience bulletin information.

4. Through adherence to facility and equipment design practices to reduce radiation levels and maintenance. BVPS-2 complies with Regulatory Guide 8.8, Revision 3, Section C.2 which recommends design features identified by experiences at operating stations. These practices are governed by Regulatory Guide 8.8 which identifies specific design features that should be considered in the engineering design to limit occupational radiation exposure to ALARA. Details are presented in Section 12.3.1.

12.1.2.2 Design Guidance

Equipment and plant design considerations support the ALARA concept by reducing the need for maintenance, BVPS-2 radiation levels, and time spent where maintenance and other operational activities are required. These considerations include:

1. Providing reliable and durable equipment and components,

2. Providing corrosion-resistant materials,

3. Providing redundancy of equipment,

4. Providing sufficient access space to equipment and piping,

5. Providing remote or mechanical devices to operate, repair, service, monitor, or inspect equipment,

6. Providing provisions for draining, flushing, and cleaning of equipment and piping which contain radioactive material,

7. Minimizing the buildup of radioactive materials in the system by continuous purification and chemistry control,

8. Providing adequate space around equipment for ease of maintenance,

9. Isolating radiation sources from equipment to be serviced by permanent and portable shielding,

10. Using remote handling devices,

11. Providing equipment arrangement and design that permits quick removal of items to be serviced,

12. Providing design features such as sealed valves, welded connections, and hard-pipe to valve stem leakoff connections,

13. Ventilation flow from low to high radiation zones, and

14. Physical barriers such as gates for controlling access to high radiation zones.

Designer guidance was provided with standards which ensured that the overall plant design met acceptable engineering standards. Guidelines were provided to ensure that the design review was in accordance with Regulatory Guide 8.8.

12.1.2.3 ALARA Design Review

In addition to the expertise in radiation reduction applied by the architect/engineer, the applicants' personnel were involved with the operation of Shippingport Atomic Power Station since 1957 and of Beaver Valley Power Station - Unit 1 since 1976. Thus, a great deal of actual experience in solving radiation design problems and techniques, went into the design review of BVPS-2.

12.1.3 Operational Considerations

The Beaver Valley Power Station radiation protection procedures establish criteria for maintaining radiation exposure ALARA during operation of BVPS-2. Additionally, the manual includes detailed procedures specifying administrative control of the program, the requirements for ALARA review, methods to be utilized in reducing radiation exposure to personnel, and documentation requirements.

12.1.3.1 ALARA Procedures

To assure that adequate emphasis is placed on the necessity to minimize personnel exposure, ALARA procedures will be incorporated in the Radiation Protection Program. These procedures implement considerations of such topics as ALARA training, ALARA review of applicable Radiological Work Permits (RWP), worker feedback, and special task training and evaluation of proposed changes in applicable facilities or equipment. The ALARA procedures provide the necessary basis for instruction of station personnel.

12.1.3.2 Organization

A description of the radiation protection organization and the responsibilities of each position is contained in Section 12.5.1. With regard to the institution of ALARA policies, a radiation protection manager is responsible for the development, coordination, maintenance, and implementation of the BVPS ALARA program, via assignment of appropriate supervisory personnel who will routinely interface with first line supervisors in planning work in radiation areas and in post job review of the work performed.

12.1.3.3 Operating Experience

The Radiological Work Permit system described in section 12.5.3.1.1 provides a mechanism for the collection and evaluation of data relating to personnel exposure. This information is collated by systems and/or components and job function, and will assist in evaluating design or procedure changes intended to minimize future radiation exposure.

12.1.3.4 Exposure Reduction

Specific exposure reduction techniques that will be employed at BVPS-2 are described in Section 12.5.3.4. Procedures will assure that applicable station activities are completed with adequate preparation and planning; work is performed with appropriate radiation protection recommendations and support; and results of post job data evaluations are applied to implement improvements.

In addition, all responsible management will at all times be vigilant for means to reduce exposure by considering employee suggestions, evaluating origins of plant exposure and investigating unusual exposure. The radiation protection staff shall ensure that adequate radiological control supplies and instrumentation are available.

Periodic management reviews are conducted for all Beaver Valley Power Station programs to ensure that workers are receiving adequate instruction in ALARA and radiation protection requirements. Implementation of the Radiation Protection Program, selected procedures, and past exposure records will also be reviewed. Management will perform formal reviews of the Beaver Valley Power Station Radiation Protection Program.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

The radiation source terms used for shield design analyses are based upon full power operation, shutdown conditions, and accident conditions. Normal operation source locations are shown on Figures 12.3-1, 12.3-2, 12.3-3, 12.3-4, 12.3-5 and 12.3-6.

The radiation sources and associated input parameters, assumptions and methodology described in this section represent those used to establish the original shielding design and are retained for historical purposes. In accordance with the original design basis, the radiation sources in the core, coolant and downstream process and effluent systems presented herein are based on a core power level of 2766 MWt, a one year fuel cycle length and 1% fuel element defects.

Core power uprate to 2900 MWt will increase the isotopic inventory in plant radioactive fluids by approximately the percentage increase in the core power level relative to that used to establish original design. The inventory of long lived isotopes will exhibit an additional increase due to the increase in fuel cycle length. The assessment of impact of power uprate on adequacy of existing plant shielding and on radwaste effluents is discussed in Section 12.3 and Sections 11.2 through 11.4, respectively.

Radiation source terms used in Chapter 15 dose analyses, and the bases for those source terms may differ from the shielding design values herein.

The sources of radioactivity contained in the streams of the various radioactive waste management systems are the nuclides generated in the reactor core, the activation of nuclides in the reactor coolant system (RCS) and the air surrounding the reactor vessel. Tables 11.1-3 and 12.2-1 present the principal parameters which were used in establishing normal operation design radiation source inventories in support of the original license. Locations for equipment containing all major sources in normal operation are outlined in Table 12.2-2.

The reactor core source description is similar to that given in Section 4.1.1 of Topical Report RP-8A (SWEC 1975).

The activity of a spent fuel assembly is calculated using appropriate fission yields, decay constants, and thermal neutron cross-sections. Isotopic inventories are based on full power operation. The core inventory at shutdown and 100 hours after shutdown based on a core power level of 2766 MWt and a one year fuel cycle length is given in Table 12.2-3. The corresponding source strength in MeV/sec, assuming a radial peaking factor of 1.65 for one fuel assembly, is given in Table 12.2-4. The above inventories were used in shielding and accident analyses supporting the original license and are retained here for historical purposes.

The core inventory at shutdown based on the uprated core power level of 2918 MWt and an 18 month fuel cycle length is presented in Table 15.0-7a. Table 15.7-6a lists the corresponding noble gas, halogen and alkali metal core inventory at 100 hours after shutdown.

The primary and secondary side system inventories based on a core power level of 2766 MWt and a one year fuel cycle length are given in Tables 11.1-2, 11.1-6, and 11.1-7. Based on selected data in Topical Report RP-8A (SWEC 1975), source strengths for various auxiliary systems are presented in Tables 12.2-5, 12.2-6, 12.2-7, 12.2-8 and 12.2-9. The above inventories were used in shielding analyses supporting the original license and are retained for historical purposes.

The design primary and secondary side system inventories based on a core power level of 2918 MWt and an 18 month fuel cycle length are given in Table 15.0-8b.

Sources used in the evaluation of equipment qualification and post-accident access doses are determined using NUREG-0737 (USNRC 1980) values for fractional releases of the core inventory which is given in Table 12.2-3. The specific activities for the contained accident sources are given for various times from T=0 to T=6 months after the accident in Tables 12.2-10, 12.2-11, 12.2-12, 12.2-13 and 12.2-14. The above inventories were used in the evaluation of equipment qualification and post-accident access doses supporting the original license and are retained here for historical purposes.

As noted in Section 3.11, the current, ongoing program of environmental qualification for electrical equipment is in accordance with the provisions of 10CFR50.49 and is implemented at BVPS-2 through several administrative procedures. The impact of power uprate on equipment qualification has been incorporated accordingly.

The assessment of impact of power uprate on post-accident access doses is discussed in Section 12.3.

A discussion of systems which contain major sources of radiation follows.

12.2.1.1 Sources for Normal Full Power Operation Shield Design

The main sources of activity during normal full power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products.

Each BVPS-2 system is shielded according to the amount of activity present and adjacent zoning and access criteria. The systems which are the major contributors to radiation levels in the plant are:

1. Reactor coolant system (RCS)

2. Chemical and volume control system (CVCS),

3. Boron recovery system (BRS),

4. Fuel pool cooling and cleanup system,

5. Gaseous, liquid, and solid waste systems, and

6. Incore instrumentation system.

12.2.1.1.1 Reactor Coolant System

Major sources in the RCS are the hot and cold leg reactor coolant piping, the pressurizer, and the steam generators. All components are located in the containment structure. Table 12.2-5 presents the respective radiation source terms.

The N-16 activity in the coolant is the controlling radiation source in the design of the RCS secondary shielding and is given in Table 12.2-15 as a function of transport time in a reactor coolant loop.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.1.2 Chemical and Volume Control System

One of the purposes of the CVCS (Section 9.3.4) is to provide continuous purification of the reactor coolant water. The major equipment items include:

1. Regenerative and excess letdown heat exchangers,

2. Mixed bed and cation bed demineralizers,

3. Reactor coolant filter,

4. Volume control tank,

5. Charging pumps, and

6. Seal water subsystem.

The seal water subsystem for the reactor coolant pumps includes the injection and return filters, and the seal water heat exchanger.

The regenerative heat exchanger and the excess letdown heat exchanger are located within the containment. The regenerative heat exchanger, combined with the letdown heat exchanger, provides for the initial cooling of the reactor coolant letdown. The excess letdown heat exchanger provides initial cooling for the reactor coolant letdown during start-up and shutdown, and when the letdown heat exchanger is out of service.

The N-16 activity dominates the dose contribution from the regenerative heat exchanger and the excess letdown heat exchanger during normal power operation. Table 12.2-6 presents the regenerative heat exchanger and excess letdown heat exchanger radiation source terms.

The balance of the CVCS equipment is located in the auxiliary building where N-16 is not a significant problem due to its short half-life and the travel time from the reactor core. Radiation source terms for the components located in the auxiliary building are given in Table 12.2-7.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.1.3 Boron Recovery System

The BRS (Section 9.3.4.6) selectively removes gases from the reactor coolant letdown flow. The BRS provides facilities for the degasification, cesium removal, and filtration of reactor coolant letdown.

Principal components of the BRS are:

1. Degasifiers and associated heat exchangers and pumps,

2. Coolant recovery filters, and

3. Cesium removal ion exchanger.

All components are located in the auxiliary building. Source terms are listed in Table 12.2-7.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.1.1.4 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system (Section 9.1.3) removes heat and radioactive impurities generated by the stored spent fuel assemblies. The cleanup system reduces radiation levels in the working area contiguous to the spent fuel storage pool and maintains the optical clarity in the spent fuel storage pool and the refueling cavity.

The expected and design concentrations in the fuel pool water are given in Tables 12.2-16 and 12.2-17, respectively. The expected and design primary coolant activities are assumed to be homogeneously mixed with the water of the refueling cavity, fuel transfer canal, and spent fuel pool. Primary coolant activities are based on 40 hours of cleanup immediately following reactor shutdown.

Components that contain major radiation source terms are the fuel pool demineralizer and the fuel pool filters which are located in the auxiliary building. Radiation source terms are presented in Table 12.2-7.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.1.6 Liquid, Gaseous, and Solid Waste Systems

These systems are designed to collect, process, and store liquid, gaseous, and solid wastes, as described in Sections 11.2, 11.3, and 11.4.

Major components of the gaseous waste system are the charcoal delay beds, the overhead gas compressors, the gaseous waste storage tank, and the waste gas surge tank. This equipment is located in the auxiliary building, except for the storage tanks which are located in the gaseous waste storage vault. Radiation source terms for these components are presented in Table 12.2-7.

Components of the liquid waste systems include the liquid waste tanks, the steam generator blowdown (SGB) hold tanks, test tanks, and evaporators. These evaporators will be used as liquid waste evaporators. All components, except the SGB evaporators and SGB hold tanks, are located in the auxiliary building. Radiation source terms are presented in Table 12.2-7. The SGB evaporators and hold tanks are located in the waste handling building. Radiation source terms are presented in Table 12.2-9.

Major radioactive sources in the solid waste system include spent resins and evaporator bottoms. Components include the spent resin hold tank, resin decant tank, resin sludge tank, and spent resin transfer pumps. All equipment, except the spent resin transfer pumps, is located in the condensate polishing building. Radiation source terms are presented in Table 12.2-8.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.1.7 Incore Instrumentation System

The incore instrumentation system (Section 4.4.6) provides information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. The incore detectors and portions of the incore detector drive cables will become activated during use.

All components of this system are located within the containment. Source terms for the neutron flux detectors and detector cables without decay are presented in Table 12.2-18. Source terms for detector drive cables with decay are presented in Table 12.2-19.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.2 Sources for Shutdown Conditions

In the reactor shutdown condition, the only additional source of radiation requiring shielding is the residual heat removal (RHR) system (Section 5.4.7), and the excess letdown heat exchanger. The excess letdown heat exchanger is discussed in Section 12.2.1.1.2.

The maximum specific source strengths in the RHR loop are given in Table 12.2-20. The RHR loop is placed in operation approximately 4 hours after reactor shutdown and reduces the reactor coolant temperature to approximately 120°F within 20 hours after shutdown. The sources are maximum values with credit taken for 4 hours of activity decay and purification.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.1.3 Sources for Design Basis Accident

The loss-of-coolant accident (LOCA) is the design basis accident (DBA) used to develop the radiation environment for equipment qualification, the radiation protection program, the post-accident access study, and shield design. The post-LOCA radiation sources of importance are listed in Table 12.2-21, along with the fractions of core activity mixed in the respective fluids. The core inventory is presented in Table 12.2-3. Equipment and systems containing these fluids are described in the following sections. Additional post-accident sources applicable to the Beaver Valley Power Station - Unit 2 (BVPS-2) design are described in Section 12.3.2.10.

As indicated earlier, Table 12.2-3 was developed in support of the original license and is considered historical.

12.2.1.3.1 Sump Water

Sump water is a radiation source both inside and outside containment. The equipment outside containment containing sump water is the safety injection system piping and pumps; the recirculation spray system piping, coolers, and pumps; and the charging pumps and associated piping. The sump water source terms at various times from T=0 to T=6 months after LOCA are given in Table 12.2-10.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.1.3.2 Primary Coolant

Primary coolant is a radiation source in the pressurizer, primary coolant lines, primary side of the steam generators' emergency coolant system, and as an airborne source from leakage of these components. The primary coolant source terms at various times from T=0 to T=6 months after LOCA are given in Table 12.2-11.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.1.3.3 Post-Accident Sample System

License Amendment 123 eliminated the commitment to have and maintain a Post Accident Sampling System (PASS), and utilize this system in the short term following an accident. This evaluation of the system as a potential radiation source is being maintained because the system has not been removed.

Sample water from the primary coolant system is depressurized and cooled. Therefore, the primary coolant specific activities shown in Table 12.2-11 are multiplied by the appropriate density correction factor to obtain the sample water specific activities. Sample system water is contained in piping and equipment in the post-accident sample area. The sample water source terms at various times from T=0 to T=6 months after LOCA are given in Table 12.2-12.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.1.3.4 Containment Atmosphere

Containment atmosphere is a radiation source within the containment building, and a radiation source to areas outside the containment through: 1) containment penetrations such as the personnel hatch and main steam line penetrations, and 2) containment atmosphere shine through the containment dome. The containment atmosphere source terms at various times from T=0 to T=6 months after LOCA are given in Table 12.2-13.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.1.3.5 Containment Atmosphere In Systems Outside Containment

Specific components and equipment which contain the containment atmosphere and are located outside containment include containment vacuum leakage monitoring piping, hydrogen control system piping, hydrogen analyzers and associated piping, and the containment purge blower. The containment atmosphere source terms (outside containment) at various times from T=0 to T=6 months after LOCA are given in Table 12.2-14.

As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

12.2.2 Airborne Radioactive Material Sources

The radioactive nuclides contained in the reactor coolant system and in the air surrounding the reactor vessel are the principal sources of airborne radionuclides. Sources derived from reactor coolant which leaks into BVPS-2 buildings result in the release of airborne contamination. The airborne radioactivity sources, which contribute to BVPS-2 effluent releases through both the radioactive waste management system and the BVPS-2 ventilation system, are described in Chapter 11.

Radioactivity leakage rates into the containment structure, turbine building, and auxiliary building are listed in Table 12.2-22. The expected case assumptions used in the calculation of radioactive airborne concentrations (Table 12.2-23) are, where applicable, taken from NUREG-0017 (USNRC 1976) and include normal operation and anticipated operational occurrences. The design case assumptions in Table 12.2-23 represent short term degradation in systems or in-system performance which is not expected to persist for extended periods of time. Airborne radioactivity at other locations is negligible.

The design and expected radioactive airborne concentrations inside the containment structure, turbine building, and auxiliary building are listed in Table 12.2-24.

The models used in calculating the radioactive airborne concentrations in the various buildings are given as follows. However, airborne levels in general access areas of the auxiliary and turbine buildings are expected to be significantly lower than the airborne levels in the equipment cubicles of these buildings. This occurs because ventilation flow paths are normally directed from areas with less potential for contamination to areas with greater potential for contamination. Activity released from relief valve venting in auxiliary building cubicles is precluded from reaching general access areas because of the ventilation system.

As indicated earlier, the tables referenced above were developed in support of the original license and are considered historical.

12.2.2.1 Containment Structure

The containment structure is not normally occupied. Radiological control procedures to control access are discussed in Section 12.5.

After shutdown, the containment purge air system further reduces the airborne activity within the containment structure. Sources from the reactor vessel head removal in preparation for refueling operations are expected to be negligible due to evacuation of the dead air space in the vessel head prior to its removal.

Radioactivity associated with primary coolant leakage is distributed uniformly in the containment atmosphere by the containment atmosphere recirculation system. Removal occurs through radioactive decay, and purging.

The containment atmosphere tritium concentration is calculated assuming the atmospheric water vapor has the same concentration (μCi of tritium per gram of water) as the reactor coolant leakage.

12.2.2.2 Turbine Building

For the purposes of calculation, it is assumed that radioactivity associated with steam leakage is uniformly distributed by the turbine building ventilation system (Section 9.4.4). Removal occurs through decay and ventilation exhaust. The tritium concentration in the turbine building is calculated assuming that all the steam leakage into the turbine building remains gaseous. Resulting airborne concentrations of radionuclides are negligible.

12.2.2.3 Auxiliary Building

For the purpose of calculation, it is assumed that radioactivity associated with primary coolant leakage is uniformly distributed by the auxiliary building ventilation system (Section 9.4.3). Removal occurs through decay and ventilation exhaust.

The tritium concentration in the auxiliary building atmosphere is conservatively calculated assuming all the primary coolant leakage into the auxiliary building evaporates. But, in fact, the leakage is collected in sumps and drains and is not generally available for evaporation.

12.2.2.4 Fuel Building

The fuel building ventilation system is described in Section 9.4.2.

Non-tritium fuel building activity is negligible because the primary system is degassed for 40 hours prior to removal of the vessel head. The tritium concentration in the fuel building atmosphere is based on a yearly average release rate into the fuel building due to evaporation.

Expected and design fuel pool concentrations are given in Tables 12.2-16 and 12.2-17, respectively. The fuel building tritium airborne concentration is 3.1×10^{-6} $\mu Ci/cm^3$ expected and 7.6×10^{-6} $\mu Ci/cm^3$ design.

As indicated earlier, the tables and concentration information referenced above were developed in support of the original license and are considered historical.

12.2.3 References for Section 12.2

Stone & Webster Engineering Corporation (SWEC) 1975. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants. Topical Report RP-8A.

U.S. Nuclear Regulatory Commission (USNRC) 1976. Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017.

USNRC 1980. Clarification of TMI Action Plan Requirements. NUREG-0737.

Tables for Section 12.2

TABLE 12.2-1 [HISTORICAL]

SELECTED PARAMETERS USED IN CALCULATION OF DESIGN RADIATION SOURCE INVENTORIES

Characteristic	Parameters
Power level (MWt)	2,766
Failed fuel fraction *	0.01
Primary to secondary leak rate (gpd)	144
Reactor operating time (days)	650

NOTE:

* Failed fuel fraction of 0.01 indicates the failure of the fuel which produces 1 percent of the reactor power.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-2

EQUIPMENT LOCATIONS

Source	UFSAR Section	Equipment Location Figure
Core	4	12.3-1
Regenerative heat exchanger	9.3.4	12.3-2
Excess letdown heat exchanger	9.3.4	12.3-2
RHR heat exchanger A & B	5.4.7	12.3-2
Boric acid filter	9.3.4	12.3-2
Fuel pool ion exchanger	9.1.3	12.3-2
Fuel pool filter A & B	9.1.3	12.3-2
Coolant recovery filter A & B	9.3.4.6	12.3-2
Cesium demineralizer A & B	9.3.4.6	12.3-2
Mixed bed demineralizer A & B	9.3.4	12.3-2
Cation bed demineralizer	9.3.4	12.3-2
Deborating demineralizer A & B	9.3.4	12.3-2
Seal water filter	9.3.4	12.3-2
Seal water injection filter A & B	9.3.4	12.3-2
Reactor coolant filter	9.3.4	12.3-2
Volume control tank	9.3.4	12.3-4
Boric acid tank A & B	9.3.4	12.3-4
Boric acid transfer pump A & B	9.3.4	12.3-4
Liquid waste tank A & B	11.2	12.3-2
Degasifier recirculating pump A & B	9.3.4.6	12.3-2
Waste gas surge tank	11.3	12.3-2
Liquid waste pump A & B	11.2	12.3-2
Seal water heat exchanger	9.3.4	12.3-3
Letdown heat exchanger	9.3.4	12.3-3
Charging pump A, B, C	9.3.4	12.3-3
Degasifier A & B	9.3.4.6	12.3-3
Waste gas charcoal delay beds	11.3	12.3-3
Air ejector charcoal delay beds	11.3	12.3-4
Primary drains transfer tank	9.3.3	12.3-2
Condensate demineralizer (A, B, C, D, E)	10.4.6	12.3-4
Condensate backwash hold tank	10.4.6	12.3-2
Condensate backwash feed tank	10.4.6	12.3-2
Resin dewatering filter (A & B)	10.4.6	12.3-5
Resin sludge tank	10.4.6	12.3-4
Resin hold tank	10.4.6	12.3-4
Resin decant tank	10.4.6	12.3-3
Steam generator blowdown/ liquid waste evaporators	11.2	12.3-2
Steam generator blowdown/ liquid waste hold tanks	11.2	12.3-3

TABLE 12.2-2 (Cont)

Source	UFSAR Section	Equipment Location Figure
Evaporator bottoms hold tank	11.2	12.3-2
Overhead gas compressors	11.3	12.3-2
Gaseous waste storage tanks	11.3	12.3-2
Cleanup filter	11.2	12.3-2

TABLE 12.2-3 [HISTORICAL]

CORE INVENTORY

Nuclide	0-Hour Decay (μCi)	100-Hour Decay (μCi)
Kr-83m	$0.12x10^{14}$	$0.16x10^{2}$
Kr-85m	$0.30x10^{14}$	$0.43x10^{7}$
Kr-85	$0.68x10^{12}$	$0.68x10^{12}$
Kr-87	$0.59x10^{14}$	*
Kr-88	$0.83x10^{14}$	$0.13x10^{4}$
Kr-89	$0.11x10^{15}$	*
Xe-131m	$0.42x10^{12}$	$0.40x10^{12}$
Xe-133m	$0.37x10^{13}$	$0.17x10^{13}$
Xe-133	$0.16x10^{15}$	$0.11x10^{15}$
Xe-135m	$0.42x10^{14}$	$0.14x10^{10}$
Xe-135	$0.41x10^{14}$	$0.21x10^{12}$
Xe-137	$0.14x10^{15}$	*
Xe-138	$0.14x10^{15}$	*
I-129	$0.17x10^{7}$	$0.17x10^{7}$
I-131	$0.69x10^{14}$	$0.50x10^{14}$
I-132	$0.99x10^{14}$	$0.42x10^{14}$
I-133	$0.16x10^{15}$	$0.55x10^{13}$
I-134	$0.18x10^{15}$	*
I-135	$0.14x10^{15}$	$0.45x10^{10}$
I-136	$0.72x10^{14}$	*
Br-83	$0.12x10^{14}$	$0.35x10^{1}$
Br-84	$0.22x10^{14}$	*
Br-85	$0.30x10^{14}$	*
Br-87	$0.58x10^{14}$	*
Se-81	$0.42x10^{13}$	*
Se-83	$0.51x10^{13}$	*
Se-84	$0.22x10^{14}$	*
Rb-88	$0.83x10^{14}$	$0.15x10^{4}$
Rb-89	$0.11x10^{15}$	*
Rb-90	$0.14x10^{15}$	*
Rb-91	$0.13x10^{15}$	*
Rb-92	$0.10x10^{15}$	*
Sr-89	$0.11x10^{15}$	$0.10x10^{15}$
Sr-90	$0.55x10^{13}$	$0.55x10^{13}$
Sr-91	$0.14x10^{15}$	$0.10x10^{12}$
Sr-92	$0.12x10^{15}$	$0.79x10^{3}$
Sr-93	$0.14x10^{15}$	*
Sr-94	$0.11x10^{15}$	*

TABLE 12.2-3 (Cont)

Nuclide	0-Hour Decay (μCi)	100-Hour Decay (μCi)
Y-90	$0.54x10^{13}$	$0.55x10^{13}$
Y-91m	$0.79x10^{14}$	$0.46x10^{11}$
Y-91	$0.14x10^{15}$	$0.13x10^{15}$
Y-92	$0.14x10^{15}$	$0.16x10^{7}$
Y-93	$0.14x10^{15}$	$0.16x10^{12}$
Y-94	$0.13x10^{15}$	*
Y-95	$0.14x10^{15}$	*
Zr-95	$0.14x10^{15}$	$0.14x10^{15}$
Zr-97	$0.14x10^{15}$	$0.22x10^{13}$
Nb-95m	$0.29x10^{13}$	$0.28x10^{13}$
Nb-95	$0.15x10^{15}$	$0.15x10^{15}$
Nb-97m	$0.13x10^{15}$	$0.21x10^{13}$
Nb-97	$0.14x10^{15}$	$0.24x10^{13}$
Mo-99	$0.14x10^{15}$	$0.51x10^{14}$
Mo-101	$0.12x10^{15}$	*
Mo-102	$0.97x10^{14}$	*
Mo-105	$0.21x10^{14}$	*
Tc-99m	$0.13x10^{15}$	$0.48x10^{14}$
Tc-101	$0.23x10^{15}$	*
Tc-102	$0.97x10^{14}$	*
Tc-105	$0.28x10^{14}$	*
Ru-103	$0.70x10^{14}$	$0.65x10^{14}$
Ru-105	$0.21x10^{14}$	$0.34x10^{7}$
Ru-106	$0.65x10^{13}$	$0.65x10^{13}$
Ru-107	$0.44x10^{13}$	*
Rh-103m	$0.70x10^{14}$	$0.65x10^{14}$
Rh-105m	$0.21x10^{14}$	$0.34x10^{7}$
Rh-105	$0.21x10^{14}$	$0.30x10^{13}$
Rh-106	$0.65x10^{13}$	$0.65x10^{13}$
Rh-107	$0.44x10^{13}$	*
Sn-127	$0.26x10^{13}$	*
Sn-128	$0.86x10^{13}$	*
Sn-130	$0.26x10^{14}$	*
Sb-127	$0.32x10^{13}$	$0.15x10^{13}$
Sb-128	$0.12x10^{13}$	*
Sb-129	$0.23x10^{14}$	$0.27x10^{7}$
Sb-130	$0.46x10^{14}$	*
Sb-131	$0.63x10^{14}$	*

TABLE 12.2-3 (Cont)

Nuclide	0-Hour Decay (μCi)	100-Hour Decay (μCi)
Sb-132	$0.78x10^{14}$	*
Sb-133	$0.79x10^{14}$	*
Te-127m	$0.69x10^{12}$	$0.73x10^{12}$
Te-127	$0.75x10^{12}$	$0.20x10^{13}$
Te-129m	$0.13x10^{14}$	$0.12x10^{14}$
Te-129	$0.25x10^{14}$	$0.12x10^{14}$
Te-131m	$0.10x10^{14}$	$0.10x10^{13}$
Te-131	$0.61x10^{14}$	$0.21x10^{12}$
Te-132	$0.99x10^{14}$	$0.40x10^{14}$
Te-133m	$0.11x10^{15}$	*
Te-133	$0.70x10^{14}$	*
Te-134	$0.16x10^{15}$	*
Cs-137	$0.58x10^{13}$	$0.58x10^{13}$
Cs-138	$0.16x10^{15}$	*
Cs-139	$0.15x10^{15}$	*
Cs-140	$0.14x10^{15}$	*
Cs-142	$0.77x10^{14}$	*
Ba-137m	$0.54x10^{13}$	$0.53x10^{13}$
Ba-139	$0.15x10^{15}$	*
Ba-140	$0.15x10^{15}$	$0.12x10^{15}$
Ba-141	$0.14x10^{15}$	*
Ba-142	$0.13x10^{15}$	*
La-140	$0.15x10^{15}$	$0.13x10^{15}$
La-141	$0.14x10^{15}$	$0.26x10^{7}$
La-142	$0.14x10^{15}$	*
La-143	$0.14x10^{15}$	*
Ce-141	$0.14x10^{15}$	$0.13x10^{15}$
Ce-143	$0.14x10^{15}$	$0.17x10^{14}$
Ce-144	$0.99x10^{14}$	$0.98x10^{14}$
Ce-145	$0.90x10^{14}$	*
Ce-146	$0.67x10^{14}$	*
Pr-143	$0.14x10^{15}$	$0.12x10^{15}$
Pr-144	$0.99x10^{14}$	$0.98x10^{14}$
Pr-145	$0.90x10^{14}$	$0.84x10^{9}$
Pr-146	$0.68x10^{14}$	*
Nd-147	$0.51x10^{14}$	$0.39x10^{14}$
Nd-149	$0.24x10^{14}$	*
Nd-151	$0.97x10^{13}$	*

TABLE 12.2-3 (Cont)

Nuclide	0-Hour Decay (μCi)	100-Hour Decay (μCi)
Pm-147	$0.17x10^{14}$	$0.17x10^{14}$
Pm-149	$0.24x10^{14}$	$0.67x10^{13}$
Pm-151	$0.97x10^{13}$	$0.85x10^{12}$
Sm-151	$0.77x10^{10}$	$0.80x10^{10}$
Sm-153	$0.36x10^{13}$	$0.81x10^{12}$

NOTE:

*Less than 1.0 microcurie.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-4 [HISTORICAL]

SOURCE STRENGTH OF A FUEL ASSEMBLY WITH
RADIAL PEAKING FACTOR OF 1.65

Energy Group (MeV)	0-Hour Decay (MeV/sec)	100-Hour Decay (MeV/sec)
0.4	1.7×10^{17}	2.6×10^{16}
0.8	7.6×10^{17}	1.6×10^{17}
1.3	7.4×10^{17}	1.2×10^{16}
1.7	6.1×10^{17}	8.3×10^{16}
2.2	1.4×10^{17}	2.4×10^{15}
2.5	2.2×10^{17}	6.0×10^{15}
3.5	5.3×10^{17}	1.6×10^{14}

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-5 [HISTORICAL]

RADIATION SOURCE TERMS,
CONTAINMENT STRUCTURE

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV)								Source Volume
	0.40	0.80	1.30	1.70	2.20	2.50	3.50	6.10	cm^3
Steam generator*	$3.6x10^1$	$1.6x10^2$	$4.3x10^1$	$2.1x10^1$	$4.1x10^0$	$1.9x10^0$	$1.7x10^0$	--	$5.86x10^7$
Reactor coolant piping	$9.5x10^4$	$2.3x10^5$	$1.0x10^5$	$1.0x10^5$	$8.6x10^4$	$1.1x10^5$	$2.5x10^4$	$1.4x10^7$	$2.65x10^8$**
Pressurizer (liquid)	$1.5x10^4$	$4.8x10^4$	$8.0x10^3$	$3.3x10^3$	$5.0x10^2$	$6.9x10^1$	$9.1x10^1$	$1.8x10^5$	$2.31x10^7$
(gas)	$1.3x10^6$	$1.7x10^2$	$1.9x10^2$	$1.1x10^3$	$3.4x10^3$	$5.0x10^3$	--	--	$1.65x10^7$
Reactor core	$5.9x10^{11}$	$2.7x10^{12}$	$2.9x10^{12}$	$2.3x10^{12}$	$5.4x10^{11}$	$8.0x10^{11}$	$2.1x10^{12}$	--	$2.63x10^7$

NOTES:

*The source intensities listed are for the shell side of the steam generator. The source intensities for the primary side are the same as those for the reactor coolant piping.

**This is the volume of the total reactor coolant.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-6 [HISTORICAL]

REGENERATIVE HEAT EXCHANGER AND
EXCESS LETDOWN HEAT EXCHANGER

Gamma Energy MeV	Regenerative Shell (9.0 ft^3) Excess Letdown Tubes (1.0 ft^3)	Regenerative Tubes (3.0 ft^3)
	Specific Source Strength (MeV/g-sec)	
0.1	$3.2x10^5$	$3.2x10^5$
0.4	$1.5x10^5$	$7.2x10^4$
0.8	$2.6x10^5$	$4.3x10^4$
1.3	$1.4x10^5$	$1.7x10^4$
1.7	$1.2x10^5$	$2.6x10^4$
2.2	$1.9x10^5$	$3.9x10^4$
2.5	$1.7x10^5$	$9.2x10^4$
3.5	$1.9x10^4$	$1.9x10^3$
6.1	$2.2x10^6$	-
7.1	$1.8x10^5$	-

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-7 [HISTORICAL]

RADIATION SOURCE TERMS
AUXILIARY BUILDING

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV)							Source Volume (cm^3)
	0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Boric acid filter	$1.4x10^5$	$4.7x10^5$	$9.9x10^3$	$7.2x10^3$	$1.5x10^3$	$8.2x10^2$	$5.9x10^2$	$4.90x10^4$
Fuel pool ion exchanger	$4.8x10^4$	$4.4x10^7$	$7.0x10^6$	--	$2.0x10^5$	$6.4x10^3$	$1.3x10^3$	$1.42x10^5$
Fuel pool filter A & B	$2.7x10^2$	$2.5x10^5$	$4.0x10^4$	--	$1.1x10^3$	$4.9x10^1$	$7.3x10^0$	$2.50x10^5$
Coolant recovery filter A & B	$2.5x10^5$	$1.8x10^7$	$1.4x10^4$	$8.2x10^3$	$1.6x10^3$	$1.6x10^3$	$1.1x10^3$	$7.08x10^4$
Cesium demineralizer A & B	$1.8x10^6$	$1.3x10^8$	$1.0x10^5$	$6.0x10^4$	$1.2x10^4$	$1.2x10^4$	$7.8x10^3$	$9.70x10^5$
Mixed bed demineralizer A & B	$1.2x10^8$	$2.8x10^8$	$3.3x10^8$	$1.2x10^7$	$3.7x10^6$	$2.5x10^5$	$1.8x10^5$	$8.50x10^5$
Cation bed demineralizer	$2.1x10^5$	$2.7x10^8$	$8.9x10^6$	--	--	--	--	$5.66x10^5$
Deborating demineralizer A & B	$6.5x10^5$	$3.1x10^5$	$9.4x10^4$	$5.9x10^4$	$8.4x10^4$	$8.5x10^4$	--	$1.22x10^6$
Seal water filter	$5.0x10^5$	$5.7x10^7$	$1.6x10^7$	$1.3x10^5$	$2.1x10^5$	$1.9x10^5$	$2.1x10^5$	$1.78x10^5$
Seal water injection filter A & B	$5.0x10^5$	$1.0x10^8$	$2.6x10^7$	$1.3x10^5$	$2.1x10^5$	$1.9x10^5$	$2.1x10^5$	$2.89x10^4$
Reactor coolant filter	$3.0x10^7$	$6.5x10^7$	$7.1x10^6$	$2.5x10^6$	$8.6x10^5$	$5.9x10^4$	$4.2x10^4$	$3.55x10^4$
Volume control tank (liquid)	$8.2x10^3$	$5.7x10^4$	$2.8x10^4$	$4.2x10^4$	$2.0x10^3$	$1.3x10^4$	$1.0x10^4$	$3.54x10^6$
(gas)	$5.6x10^3$	$1.5x10^5$	$8.2x10^{-2}$	$1.1x10^{-1}$	$2.9x10^{-1}$	$3.6x10^{-1}$	$5.9x10^{-2}$	$4.96x10^6$
Boric acid tank A & B	$2.3x10^4$	$9.7x10^5$	$4.9x10^4$	$4.6x10^2$	$1.1x10^2$	$6.0x10^1$	$1.3x10^{-2}$	$5.15x10^7$
Boric acid transfer pump A & B	$2.3 x10^4$	$9.7x10^5$	$4.9x10^4$	$4.6x10^2$	$1.1x10^2$	$6.0x10^1$	$1.3x10^{-2}$	*
Liquid waste tank A & B	$2.8x10^3$	$1.9x10^5$	$7.9x10^4$	$5.1x10^4$	$1.2x10^4$	$8.9x10^1$	$1.7x10^1$	$1.89x10^7$
Degassifier recirculating pump A & B	$8.2x10^3$	$5.7x10^4$	$2.8x10^4$	$4.2x10^4$	$2.0x10^3$	$1.3x10^4$	$1.0x10^4$	*
Waste gas surge tank	$1.3x10^4$	$1.9x10^5$	$1.5x10^3$	$1.8x10^3$	$4.9x10^3$	$6.1x10^3$	$1.1x10^3$	$1.47x10^6$
Liquid waste pump A & B	$2.8x10^3$	$1.9x10^5$	$7.9x10^4$	$5.1x10^4$	$1.2x10^4$	$8.9x10^1$	$1.7x10^1$	*

TABLE 12.2-7 (Cont)

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV) 0.40	0.80	1.30	1.70	2.20	2.50	`	Source Volume (cm^3)
Seal water heat exchanger	$5.0x10^5$	$1.0 x10^8$	$2.6 x10^7$	$1.3 x10^5$	$2.1 x10^5$	$1.9 x10^5$	$2.1 x10^5$	$1.21 x10^5$
Letdown heat exchanger	$1.5 x10^5$	$2.5 x10^5$	$1.1 x10^5$	$1.2 x10^5$	$9.5 x10^4$	$1.3 x10^5$	$3.7 x10^4$	$3.46 x10^5$
Charging pump A, B, C	$1.5 x10^5$	$2.5 x10^5$	$1.1 x10^5$	$1.2 x10^5$	$9.5 x10^4$	$1.3 x10^5$	$3.7 x10^4$	*
Degassifier A & B (liquid)	$8.2 x10^3$	$5.7 x10^4$	$2.8 x10^4$	$4.2 x10^4$	$2.0 x10^3$	$1.3 x10^4$	$1.0 x10^4$	$5.70 x10^6$
(vapor)	$1.5 x10^3$	$1.2 x10^3$	$7.4 x10^1$	$6.4 x10^2$	$1.8 x10^3$	$2.4 x10^3$	$7.8x10^1$	$7.10 x10^6$
Waste gas charcoal delay beds A	$8.1 x10^7$	$2.8 x10^6$	$1.4 x10^6$	$2.0 x10^6$	$5.2 x10^6$	$7.5 x10^6$	$1.0 x10^6$	$8.48 x10^5$
B-D	$1.6 x10^7$	$4.4 x10^4$	$3.3 x10^4$	$4.3 x10^4$	$1.2 x10^5$	$1.5 x10^5$	$2.3 x10^4$	$8.48 x10^5$
Air ejector charcoal delay beds A-L	$1.2 x10^5$	$1.5 x10^4$	$2.8 x10^3$	$2.9 x10^3$	$4.9 x10^3$	$6.8 x10^3$	$9.4 x10^2$	$1.56 x10^6$
Primary drains transfer tank	$9.4 x10^4$	$2.1 x10^5$	$9.5 x10^4$	$8.5 x10^4$	$7.1 x10^4$	$1.0 x10^5$	$1.7 x10^4$	$3.46 x10^6$
Overhead gas compressors A & B	$1.3 x10^4$	$1.9 x10^5$	$1.5 x10^3$	$1.8 x10^3$	$4.9 x10^3$	$6.1 x10^3$	$1.1 x10^3$	$1.42 x10^4$
Gaseous waste storage tanks**	$6.0 x10^5$	$2.4 x10^7$	$0.8 x10^3$	$5.3 x10^4$	$1.5 x10^5$	$2.0 x10^5$	0.0	$3.74 x10^6$
Cleanup filter	$1.5 x10^6$	$5.9 x10^7$	$4.4 x10^6$	$7.7 x10^4$	$6.2 x10^2$	$3.3 x10^3$	$1.9 x10^2$	$2.90 x10^4$

NOTES:

*Pump volumes are small compared to volume of associated piping.
**Gaseous waste storage tanks are located underground, north of fuel building in the gaseous waste storage vault.
The tank spectrum corresponds to the technical specification limit for activity assuming the total activity is in one gaseous waste storage tank.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-8 [HISTORICAL]

RADIATION SOURCE TERMS
CONDENSATE POLISHING BUILDING

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV)							Source Volume (cm^3)
	0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Condensate demineralizer (A, B, C, D and E)	$4.1x10^5$	$1.4x10^6$	$1.8x10^5$	$3.0 x10^4$	$7.6 x10^3$	$2.4 x10^3$	$3.8 x10^1$	$1.32 x10^5$
Condensate backwash receiving tank	$2.9 x10^3$	$9.7 x10^3$	$1.3 x10^3$	$2.1 x10^2$	$5.4 x10^1$	$1.7 x10^1$	$2.8 x10^{-1}$	$1.85 x10^7$
Condensate backwash feed tank	$1.6 x10^4$	$1.0 x10^6$	$4.4x10^4$	$8.2 x10^2$	$1.6 x10^2$	$6.3 x10^1$	$7.1 x10^{-1}$	$4.84 x10^5$
Resin dewatering filter (A and B)	$3.2x10^5$	$1.1 x10^6$	$1.4x10^5$	$2.3 x10^4$	$5.9 x10^3$	$1.9 x10^3$	$3.0 x10^1$	$1.7 x10^5$
Resin sludge tank	$4.1 x10^5$	$1.4 x10^6$	$1.8 x10^5$	$2.9 x10^4$	$7.6 x10^3$	$2.4 x10^3$	$3.8 x10^1$	$7.95 x10^5$
Resin hold tank	$1.7 x10^7$	$4.1 x10^7$	$1.4 x10^7$	$9.4 x10^6$	$1.9 x10^6$	$2.4 x10^4$	$2.1 x10^5$	$5.72 x10^6$
Resin decant tank	$1.9 x10^7$	$4.5 x10^7$	$1.6 x10^7$	$1.0 x10^7$	$2.1 x10^6$	$2.7 x10^4$	$2.4 x10^5$	$1.89 x10^6$

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-9 [HISTORICAL]

RADIATION SOURCE TERMS, WASTE HANDLING BUILDING

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV)							Source Volume (cm^3)
	0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Steam generator blowdown/ liquid waste evaporators	3.9×10^5	2.0×10^6	2.2×10^5	1.4×10^4	4.2×10^3	1.9×10^3	1.7×10^0	3.45×10^6
Steam generator blowdown/ liquid waste hold tanks	2.6×10^3	6.6×10^3	9.2×10^2	2.1×10^2	6.2×10^1	2.1×10^1	8.0×10^{-2}	1.89×10^8
Evaporator bottoms hold tank	3.9×10^5	2.0×10^6	2.2×10^5	1.4×10^4	4.2×10^3	1.9×10^3	1.7×10^0	8.33×10^6

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-10 [HISTORICAL]

RADIATION SOURCE TERMS, SUMP WATER POST - LOCA

Time After LOCA (hr)	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.4	0.8	1.3	1.7	2.2	2.5	3.5
0.0	1.89×10^8	3.13×10^9	2.11×10^9	1.07×10^9	2.19×10^8	2.43×10^8	1.11×10^9
0.083	1.50×10^8	3.00×10^9	1.62×10^9	1.15×10^9	1.87×10^8	1.37×10^8	1.91×10^8
1	1.30×10^8	2.02×10^9	1.24×10^9	1.02×10^9	1.57×10^8	1.42×10^8	1.82×10^8
2	1.26×10^8	1.41×10^9	9.31×10^8	6.66×10^8	1.26×10^8	5.58×10^7	1.08×10^8
8	1.21×10^8	5.79×10^8	3.69×10^8	2.55×10^8	4.65×10^7	4.39×10^6	1.80×10^7
24	1.12×10^8	3.08×10^8	6.83×10^7	7.05×10^7	7.76×10^6	2.01×10^6	3.85×10^5
96	8.51×10^7	8.29×10^7	3.22×10^6	2.32×10^7	6.79×10^5	1.65×10^6	4.50×10^4
192	6.02×10^7	5.17×10^7	1.93×10^6	1.84×10^7	3.93×10^5	1.27×10^6	3.74×10^4
360	3.34×10^7	3.84×10^7	1.07×10^6	1.25×10^7	2.29×10^5	8.38×10^5	2.64×10^4
720	9.88×10^6	2.66×10^7	4.67×10^5	5.56×10^6	1.78×10^5	3.65×10^5	1.29×10^4
2,160	4.23×10^5	1.35×10^7	6.38×10^4	2.62×10^5	1.53×10^5	1.83×10^4	2.22×10^3
4,320	1.29×10^5	5.88×10^6	2.43×10^4	3.97×10^4	1.23×10^5	3.65×10^3	1.53×10^3

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-11 [HISTORICAL]

RADIATION SOURCE TERMS, PRIMARY COOLANT POST - LOCA

Time After LOCA (hr)	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.4	0.8	1.3	1.7	2.2	2.5	3.5
0.0	6.02×10^{9}	5.53×10^{10}	2.74×10^{10}	5.64×10^{10}	3.18×10^{10}	2.05×10^{10}	3.21×10^{10}
0.083	5.50×10^{9}	5.10×10^{10}	2.12×10^{10}	3.23×10^{10}	2.66×10^{10}	1.93×10^{10}	8.49×10^{9}
1	5.01×10^{9}	3.12×10^{10}	1.62×10^{10}	1.50×10^{10}	9.03×10^{9}	1.37×10^{10}	2.34×10^{9}
2	4.75×10^{9}	2.17×10^{10}	1.22×10^{10}	1.01×10^{10}	6.15×10^{8}	9.06×10^{9}	1.38×10^{9}
8	3.89×10^{9}	8.37×10^{9}	4.79×10^{9}	3.62×10^{9}	1.55×10^{8}	1.42×10^{9}	2.31×10^{8}
24	2.78×10^{9}	4.11×10^{9}	8.78×10^{8}	9.11×10^{8}	1.17×10^{8}	4.95×10^{7}	4.94×10^{6}
96	1.53×10^{9}	1.07×10^{9}	4.13×10^{7}	2.97×10^{8}	8.71×10^{6}	2.12×10^{7}	5.77×10^{5}
192	1.03×10^{9}	6.64×10^{8}	2.48×10^{7}	2.36×10^{8}	5.05×10^{6}	1.63×10^{7}	4.79×10^{5}
360	5.34×10^{8}	4.93×10^{8}	1.37×10^{7}	1.61×10^{8}	2.94×10^{6}	1.08×10^{7}	3.39×10^{5}
720	1.44×10^{8}	3.42×10^{8}	5.99×10^{6}	7.13×10^{1}	2.29×10^{6}	4.68×10^{6}	1.65×10^{5}
2,160	5.59×10^{6}	1.74×10^{8}	8.19×10^{5}	3.36×10^{6}	1.96×10^{6}	2.34×10^{5}	2.85×10^{4}
4,320	1.65×10^{6}	7.59×10^{7}	3.12×10^{5}	5.09×10^{5}	1.57×10^{6}	4.68×10^{4}	1.96×10^{4}

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-12 [HISTORICAL]

RADIATION SOURCE TERMS, SAMPLE SYSTEM POST - LOCA

Time After LOCA (hr)	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.4	0.8	1.3	1.7	2.2	2.5	3.5
0.0	$8.08x10^{9}$	$7.42x10^{10}$	$3.68x10^{10}$	$7.57x10^{10}$	$4.27x10^{10}$	$2.75x10^{10}$	$4.31x10^{10}$
0.083	$7.38x10^{9}$	$6.84x10^{10}$	$2.84x10^{10}$	$4.33x10^{10}$	$3.57x10^{10}$	$2.59x10^{10}$	$1.14x10^{10}$
1	$6.72x10^{9}$	$4.19x10^{10}$	$2.17x10^{10}$	$2.01x10^{10}$	$1.21x10^{10}$	$1.84x10^{10}$	$3.14x10^{9}$
2	$6.37x10^{9}$	$2.91x10^{10}$	$1.64x10^{10}$	$1.36x10^{10}$	$8.25x10^{9}$	$1.22x10^{10}$	$1.85x10^{9}$
8	$5.22x10^{9}$	$1.12x10^{10}$	$6.43x10^{9}$	$4.86x10^{9}$	$2.08x10^{9}$	$1.91x10^{9}$	$3.10x10^{8}$
24	$3.73x10^{9}$	$5.52x10^{9}$	$1.18x10^{9}$	$1.22x10^{9}$	$1.57x10^{8}$	$6.64x10^{7}$	$6.63x10^{6}$
96	$2.05x10^{9}$	$1.44x10^{9}$	$5.54x10^{7}$	$3.99x10^{8}$	$1.17x10^{7}$	$2.84x10^{7}$	$7.74x10^{5}$
192	$1.38x10^{9}$	$8.91x10^{8}$	$3.33x10^{7}$	$3.17x10^{8}$	$6.78x10^{6}$	$2.19x10^{7}$	$6.43x10^{5}$
360	$7.17x10^{8}$	$6.62x10^{8}$	$1.84x10^{7}$	$2.16x10^{8}$	$3.95x10^{6}$	$1.45x10^{7}$	$4.55x10^{5}$
720	$1.93x10^{8}$	$4.59x10^{8}$	$8.04x10^{6}$	$9.57x10^{7}$	$3.07x10^{6}$	$6.28x10^{5}$	$2.21x10^{5}$
2,160	$7.50x10^{6}$	$2.33x10^{8}$	$1.10x10^{6}$	$4.51x10^{6}$	$2.63x10^{6}$	$3.14x10^{5}$	$3.82x10^{4}$
4,320	$2.21x10^{6}$	$1.02x10^{8}$	$4.19x10^{5}$	$6.83x10^{5}$	$2.11x10^{6}$	$6.28x10^{4}$	$2.63x10^{4}$

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-13 [HISTORICAL]

RADIATION SOURCE TERMS, CONTAINMENT ATMOSPHERE POST - LOCA

Time After LOCA (hr)	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.4	0.8	1.3	1.7	2.2	2.5	3.5
0.0	2.96×10^{7}	2.88×10^{8}	1.37×10^{8}	2.95×10^{8}	1.70×10^{8}	1.08×10^{8}	1.65×10^{8}
0.083	2.72×10^{7}	2.64×10^{8}	1.03×10^{8}	1.56×10^{8}	1.42×10^{8}	9.85×10^{7}	3.97×10^{7}
1	2.57×10^{7}	1.58×10^{8}	7.37×10^{7}	5.57×10^{7}	4.62×10^{7}	6.53×10^{7}	1.42×10^{6}
2	2.45×10^{7}	1.10×10^{8}	5.69×10^{7}	4.35×10^{7}	3.10×10^{7}	4.59×10^{7}	3.85×10^{5}
8	2.01×10^{7}	4.04×10^{7}	2.38×10^{7}	1.72×10^{7}	7.89×10^{6}	7.50×10^{6}	1.51×10^{2}
24	1.43×10^{7}	1.86×10^{7}	4.43×10^{6}	3.18×10^{6}	6.09×10^{5}	1.57×10^{5}	--
96	7.83×10^{6}	3.17×10^{5}	9.96×10^{4}	8.78×10^{4}	3.39×10^{4}	1.49×10^{4}	--
192	5.30×10^{6}	1.24×10^{6}	4.12×10^{4}	2.39×10^{4}	1.43×10^{4}	6.33×10^{3}	--
360	2.72×10^{6}	5.82×10^{5}	9.21×10^{3}	5.03×10^{3}	3.18×10^{3}	1.41×10^{3}	--
720	6.83×10^{5}	1.51×10^{5}	3.69×10^{2}	2.02×10^{2}	1.28×10^{2}	5.68×10^{1}	--
2,160	4.29×10^{3}	3.02×10^{3}	--	--	--	--	--
4,320	6.53×10^{0}	2.13×10^{3}	--	--	--	--	--

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-14 [HISTORICAL]

RADIATION SOURCE TERMS, CONTAINMENT ATMOSPHERE IN
SYSTEMS OUTSIDE CONTAINMENT POST - LOCA

Time After LOCA (hr)	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.4	0.8	1.3	1.7	2.2	2.5	3.5
0.0	2.96×10^{7}	2.88×10^{8}	1.37×10^{8}	2.95×10^{8}	1.70×10^{8}	1.08×10^{8}	1.65×10^{8}
0.083	2.72×10^{7}	2.64×10^{8}	1.03×10^{8}	1.56×10^{8}	1.42×10^{8}	9.85×10^{7}	3.97×10^{7}
1	2.19×10^{7}	9.31×10^{7}	3.77×10^{7}	3.31×10^{7}	4.20×10^{7}	6.49×10^{7}	7.12×10^{5}
2	2.07×10^{7}	6.44×10^{7}	2.91×10^{7}	2.58×10^{7}	2.77×10^{7}	4.54×10^{7}	1.93×10^{5}
8	1.64×10^{7}	2.27×10^{7}	1.21×10^{7}	9.52×10^{6}	6.52×10^{6}	7.41×10^{6}	7.58×10^{1}
24	1.08×10^{7}	9.73×10^{6}	2.22×10^{6}	1.61×10^{6}	3.52×10^{5}	1.42×10^{5}	--
96	5.08×10^{6}	1.59×10^{6}	4.98×10^{4}	4.39×10^{4}	1.69×10^{4}	7.45×10^{3}	--
192	3.34×10^{6}	6.20×10^{5}	2.06×10^{4}	1.20×10^{4}	7.15×10^{3}	3.16×10^{3}	--
360	1.64×10^{6}	2.92×10^{5}	4.60×10^{3}	2.52×10^{3}	1.59×10^{3}	7.06×10^{2}	--
720	3.88×10^{5}	7.66×10^{4}	1.85×10^{2}	1.01×10^{2}	6.41×10^{1}	2.84×10^{1}	--
2,160	2.58×10^{3}	2.60×10^{3}	4.83×10^{-4}	2.64×10^{-4}	1.67×10^{-4}	7.40×10^{-5}	--
4,320	5.78×10^{0}	2.13×10^{3}	--	--	--	--	--

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-15 [HISTORICAL]

REACTOR COOLANT N-16 ACTIVITY

Position in Loop	Loop Transit Time (sec)	N-16 Activity (μci/g)
Leaving core	0	132
Leaving reactor vessel	1.0	119
Entering steam generator	1.4	110
Leaving steam generator	6.6	74
Entering reactor coolant pump	7.2	70
Entering reactor vessel	7.9	65
Entering core	10.4	52
Leaving core	11.3	132

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-16 [HISTORICAL]

EXPECTED CONCENTRATIONS IN FUEL POOL WATER*

Nuclide	Concentration ($\mu Ci/cm^3$)
I-131	$9.6x10^{-5}$
I-132	$7.9x10^{-6}$
I-133	$4.2x10^{-5}$
I-135	$1.3x10^{-6}$
Rb-88	$2.1x10^{-9}$
Sr-89	$1.4x10^{-7}$
Sr-90	$4.2x10^{-9}$
Sr-91	$1.6x10^{-8}$
Y-90	$6.3x10^{-8}$
Y-91m	$1.0x10^{-8}$
Y-91	$2.7x10^{-8}$
Y-93	$9.5x10^{-10}$
Zr-95	$2.4x10^{-8}$
Nb-95	$2.0x10^{-8}$
Mo-99	$2.4x10^{-3}$
Tc-99m	$1.1x10^{-3}$
Ru-103	$1.7x10^{-8}$
Ru-106	$4.2x10^{-9}$
Rh-103m	$1.7x10^{-8}$
Rh-106	$4.2x10^{-9}$
Te-127m	$1.7x10^{-7}$
Te-127	$1.2x10^{-7}$
Te-129m	$5.5x10^{-7}$
Te-129	$3.5x10^{-7}$
Te-131m	$4.1x10^{-7}$
Te-131	$7.5x10^{-8}$
Te-132	$7.7x10^{-6}$
Cs-134	$1.1x10^{-3}$
Cs-136	$5.2x10^{-4}$
Cs-137	$7.7x10^{-4}$
Ba-137m	$7.2x10^{-4}$`
Ba-140	$8.0x10^{-8}$
La-140	$7.2x10^{-8}$
Ce-141	$2.7x10^{-8}$
Ce-143	$7.1x10^{-9}$

TABLE 12.2-16 (Cont)

Nuclide	Concentration ($\mu Ci/cm^3$)
Ce-144	$1.3x10^{-8}$
Pr-143	$1.9x10^{-8}$
Pr-144	$1.3x10^{-8}$
Cr-51	$7.3x10^{-7}$
Mn-54	$1.2x10^{-7}$
Fe-55	$6.5x10^{-7}$
Fe-59	$4.1x10^{-7}$
Co-58	$6.4x10^{-6}$
Co-60	$8.0x10^{-7}$
H-3	$9.7x10^{-2}$

NOTE:

*Nuclides with concentrations less than $1.0x10^{-10}$ $\mu Ci/cm^3$ are not included.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-17 [HISTORICAL]

DESIGN CONCENTRATIONS IN FUEL POOL WATER*

Nuclide	Concentration ($\mu Ci/cm^3$)
Br-83	$3.1x10^{-10}$
I-131	$8.3x10^{-4}$
I-132	$7.1x10^{-5}$
I-133	$4.0x10^{-4}$
I-135	$1.3x10^{-5}$
Rb-88	$3.8x10^{-8}$
Sr-89	$1.5x10^{-6}$
Sr-90	$6.2x10^{-8}$
Sr-91	$4.1x10^{-8}$
Y-90	$5.6x10^{-6}$
Y-91m	$2.7x10^{-8}$
Y-91	$2.5x10^{-7}$
Y-92	$4.2x10^{-10}$
Y-93	$8.2x10^{-9}$
Zr-95	$2.5x10^{-7}$
Zr-97	$2.9x10^{-8}$
Nb-95m	$5.1x10^{-9}$
Nb-95	$2.6x10^{-7}$
Nb-97m	$2.8x10^{-8}$
Nb-97	$3.1x10^{-8}$
Mo-99	$8.7x10^{-2}$
Tc-99m	$4.0x10^{-2}$
Ru-103	$1.2x10^{-7}$
Ru-106	$1.2x10^{-8}$
Rh-103m	$1.2x10^{-7}$
Rh-105	$1.5x10^{-8}$
Rh-106	$1.2x10^{-8}$
Sb-127	$3.9x10^{-9}$
Te-127m	$7.0x10^{-7}$
Te-127	$6.9x10^{-7}$
Te-129m	$1.4x10^{-5}$
Te-129	$8.8x10^{-6}$

TABLE 12.2-17 (Cont)

Nuclide	Concentration ($\mu Ci/cm^3$)
Te-131m	3.4×10^{-6}
Te-131	6.2×10^{-7}
Te-132	6.9×10^{-5}
Cs-134	1.1×10^{-2}
Cs-136	5.6×10^{-3}
Cs-137	6.3×10^{-2}
Ba-137m	5.9×10^{-2}
Ba-140	1.5×10^{-6}
La-140	1.0×10^{-6}
Ce-141	2.4×10^{-7}
Ce-143	8.1×10^{-8}
Ce-144	1.8×10^{-7}
Pr-143	2.4×10^{-7}
Pr-144	1.8×10^{-7}
Pr-145	5.7×10^{-10}
Nd-147	7.5×10^{-10}
Pm-147	2.7×10^{-8}
Pm-149	2.2×10^{-8}
Pm-151	4.8×10^{-9}
Sm-153	2.8×10^{-9}
Cr-51	2.2×10^{-6}
Mn-54	3.7×10^{-7}
Mn-56	2.4×10^{-10}
Fe-55	1.9×10^{-6}
Fe-59	1.2×10^{-6}
Co-58	1.9×10^{-5}
Co-60	2.4×10^{-6}
H-3	2.4×10^{-1}

NOTE:

*Nuclides with concentrations less than 1.0×10^{-10} $\mu Ci/cm^3$ are not included.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-18 [HISTORICAL]

IRRADIATED INCORE DETECTOR AND DRIVE CABLE
MAXIMUM WITHDRAWAL SOURCE STRENGTHS

	Source Intensity (MeV/cm^3-sec) for Energy (MeV)								
Source	0.40	0.80	1.30	1.70	2.20	2.50	3.50	4.50	5.50
Incore detector drive cable	6.0×10^{8}	5.1×10^{10}	1.6×10^{10}	3.1×10^{8}	3.8×10^{10}	1.3×10^{9}	1.6×10^{9}	--	--
Incore detector	3.8×10^{10}	1.6×10^{11}	1.1×10^{11}	1.1×10^{11}	2.9×10^{10}	3.1×10^{10}	3.7×10^{10}	1.5×10^{10}	1.4×10^{9}

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-19 [HISTORICAL]

IRRADIATED INCORE DETECTOR DRIVE CABLE SOURCE STRENGTHS

	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
Source	0.40	0.80	1.30	1.70	2.20	2.50	3.50
Drive Cable with decay of:							
8 hours	$5.8x10^{8}$	$1.7x10^{10}$	$1.6x10^{10}$	$2.1x10^{7}$	$4.5x10^{9}$	$1.5x10^{8}$	$2.0x10^{8}$
1 day	$5.7x10^{8}$	$1.2x10^{10}$	$1.6x10^{10}$	$1.1x10^{7}$	$6.1x10^{7}$	$2.0x10^{6}$	$2.6x10^{6}$
1 week	$4.9x10^{8}$	$1.2x10^{10}$	$1.5x10^{10}$	$1.1x10^{7}$	--	--	--
1 month	$2.8x10^{8}$	$1.1x10^{10}$	$1.2x10^{10}$	$8.5x10^{6}$	--	--	--
6 months	$9.8x10^{8}$	$7.8x10^{9}$	$5.6x10^{9}$	$2.0x10^{6}$	--	--	--
1 year	$2.8x10^{5}$	$5.0x10^{9}$	$4.2x10^{9}$	$3.3x10^{5}$	--	--	--
5 years	--	$2.0x10^{8}$	$2.5x10^{9}$	--	--	--	--

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-20 [HISTORICAL]

RADIATION SOURCE TERMS
FOR SHUTDOWN CONDITIONS
(4 HOURS ONLY)

Source	Source Intensity (MeV/cm^3-sec) for Energy (MeV)						
	0.40	0.80	1.30	1.70	2.20	2.50	3.50
RHRS heat exchangers (A & B)	5.0×10^4	9.9×10^4	3.3×10^4	2.0×10^4	1.5×10^4	1.7×10^4	2.7×10^3

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-21

POST-LOCA RADIATION SOURCES

Radiation Source	Fraction of Core Activity Released*
Sump water	50% halogens, 1% remainder
Primary coolant	100% noble gases, 50% halogens, 1% remainder
Sample system	100% noble gases, 50% halogens, 1% remainder
Containment atmosphere (inside containment)	100% noble gases, 50% halogens
Containment atmosphere	100% noble gases, 50% halogens: T≤5 min after LOCA
(outside containment)	100% noble gases, 25% halogens: T>5 min after LOCA

NOTE:

*NUREG-0737 Item II.B.2 (USNRC 1980).

TABLE 12.2-22 [HISTORICAL]

RADIOACTIVITY LEAKAGE INTO MAJOR BUILDINGS
(μCi/sec)

	Containment Structure		Auxiliary Building		Turbine Building	
Nuclide	Design	Expected	Design	Expected	Design	Expected
H-3	7.09×10^{1}	1.62×10^{1}	2.94×10^{0}	6.73×10^{-1}	5.56×10^{0}	1.72×10^{-1}
I-131	5.14×10^{-2}	5.87×10^{-3}	1.60×10^{-2}	1.83×10^{-3}	2.79×10^{-3}	2.57×10^{-5}
I-132	1.79×10^{-2}	2.23×10^{-3}	5.58×10^{-3}	6.94×10^{-4}	8.23×10^{-4}	7.50×10^{-6}
I-133	8.02×10^{-2}	8.50×10^{-3}	2.50×10^{-2}	2.65×10^{-3}	3.99×10^{-3}	3.43×10^{-5}
I-134	1.12×10^{-2}	1.09×10^{-3}	3.49×10^{-3}	3.40×10^{-4}	1.84×10^{-5}	1.48×10^{-6}
I-135	4.31×10^{-2}	4.25×10^{-3}	1.34×10^{-2}	1.32×10^{-3}	1.78×10^{-3}	1.52×10^{-5}
Kr-83m	8.79×10^{0}	4.05×10^{-1}	3.65×10^{-1}	1.68×10^{-2}	3.99×10^{-4}	1.57×10^{-6}
Kr-85m	4.29×10^{1}	1.80×10^{0}	1.78×10^{0}	7.48×10^{-2}	1.95×10^{-3}	6.86×10^{-6}
Kr-85	2.27×10^{2}	2.43×10^{0}	9.42×10^{0}	1.01×10^{-1}	1.02×10^{-2}	9.22×10^{-6}
Kr-87	2.45×10^{1}	1.26×10^{0}	1.02×10^{0}	5.21×10^{-2}	1.11×10^{-3}	4.50×10^{-6}
Kr-88	6.54×10^{1}	3.64×10^{0}	2.72×10^{0}	1.51×10^{-1}	2.96×10^{-3}	1.39×10^{-5}
Kr-89	2.07×10^{0}	1.17×10^{-1}	8.58×10^{-2}	4.88×10^{-3}	9.35×10^{-5}	4.50×10^{-7}
Xe-131m	2.12×10^{0}	1.21×10^{-1}	8.85×10^{-2}	5.04×10^{-3}	9.68×10^{-5}	4.93×10^{-7}
Xe-133m	6.30×10^{1}	8.50×10^{-1}	2.61×10^{0}	3.53×10^{-2}	2.85×10^{-3}	3.22×10^{-6}
Xe-133	5.37×10^{2}	3.44×10^{1}	2.23×10^{1}	1.43×10^{-7}	2.42×10^{-2}	1.33×10^{-4}
Xe-135m	2.23×10^{1}	3.04×10^{-1}	9.25×10^{-1}	1.26×10^{-2}	1.01×10^{-3}	1.14×10^{-6}
Xe-135	6.58×10^{1}	4.25×10^{0}	2.73×10^{0}	1.77×10^{-1}	2.98×10^{-3}	1.67×10^{-5}
Xe-137	3.34×10^{0}	2.00×10^{-1}	1.39×10^{-1}	8.32×10^{-3}	1.51×10^{-4}	8.15×10^{-7}
Xe-138	1.38×10^{1}	1.01×10^{0}	5.72×10^{-1}	4.20×10^{-2}	6.24×10^{-4}	3.86×10^{-6}

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-23 [HISTORICAL]

ASSUMPTIONS USED IN THE CALCULATION OF AIRBORNE CONCENTRATIONS

Characteristic	Containment Structure	Auxiliary Building	Turbine Building
Reactor coolant equilibrium concentrations			
Design case	Table 11.1-2	Table 11.1-2	-
Expected case	Table 11.1-2	Table 11.1-2	-
Secondary side equilibrium concentrations			
Design case	-	-	Table 11.1-6
Expected case	-	-	Table 11.1-6
Leak rate into building			
Equivalent reactor coolant leakage(lb/day)	3.853×10^3	160.0	
Equivalent main stream leakage(lb/hr)			1,700.0
Partition factor (PF)			
Noble gases	1.0	1.0	1.0
Halogens	0.001	0.0075	1.0
Mixing in building atmosphere(%)	70.0	-	-
Building ventilation rate(cfm)	3.0×10^4(during purge)	2.82×10^4	7.50×10^5
Building free volume(ft3)	1.8×10^6	1.068×10^6	3.380×10^6
Recirculation filters	yes	no	no
Recirculation filter efficiency(%)	90.0	-	-
Recirculation rate(cfm)	1.0×10^4	-	-
Number of purges per year	4	-	-
Recirculation time prior to purging(hr)	16.0	-	-
Containment maximum temperature for normal service conditions (°F)	105		
Containment maximum relative humidity for normal service conditions (percent)	60		

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.2-24 [HISTORICAL]

RADIOACTIVE AIRBORNE CONCENTRATIONS INSIDE MAJOR BUILDINGS
($\mu Ci/cm^3$)

	Containment (Before Recirculation)		Containment (After Recirculation)		Auxiliary Building		Turbine Building	
Nuclide	Design	Expected	Design	Expected	Design	Expected	Design	Expected
H-3	1.98×10^{-4}	5.65×10^{-5}	1.98×10^{-4}	5.65×10^{-5}	2.21×10^{-7}	5.05×10^{-8}	1.57×10^{-8}	4.85×10^{-10}
I-131	1.01×10^{-6}	1.16×10^{-7}	3.31×10^{-8}	3.80×10^{-9}	1.20×10^{-9}	1.37×10^{-10}	7.87×10^{-12}	7.27×10^{-14}
I-132	4.21×10^{-9}	5.23×10^{-10}	1.18×10^{-12}	1.47×10^{-13}	3.52×10^{-10}	4.37×10^{-11}	2.27×10^{-12}	2.07×10^{-14}
I-133	1.70×10^{-7}	1.81×10^{-8}	3.47×10^{-9}	3.69×10^{-10}	1.84×10^{-9}	1.95×10^{-10}	1.12×10^{-11}	9.67×10^{-14}
I-134	1.00×10^{-9}	9.78×10^{-11}	1.09×10^{-16}	1.07×10^{-17}	1.75×10^{-10}	1.70×10^{-11}	4.89×10^{-14}	3.94×10^{-15}
I-135	2.92×10^{-8}	2.87×10^{-9}	1.90×10^{-10}	1.86×10^{-11}	9.47×10^{-10}	9.34×10^{-11}	5.00×10^{-12}	4.27×10^{-14}
Kr-83m	1.65×10^{-6}	7.59×10^{-8}	3.90×10^{-9}	1.79×10^{-10}	2.22×10^{-8}	1.02×10^{-9}	1.10×10^{-12}	4.30×10^{-15}
Kr-85m	1.96×10^{-5}	8.24×10^{-7}	1.65×10^{-6}	6.92×10^{-8}	1.22×10^{-7}	5.11×10^{-9}	5.44×10^{-12}	1.92×10^{-14}
Kr-85	3.44×10^{-2}	3.69×10^{-4}	3.43×10^{-2}	3.69×10^{-4}	7.08×10^{-7}	7.58×10^{-9}	2.90×10^{-11}	2.60×10^{-14}
Kr-87	3.19×10^{-6}	1.64×10^{-7}	5.43×10^{-10}	2.74×10^{-11}	5.67×10^{-8}	2.91×10^{-9}	3.02×10^{-12}	1.22×10^{-14}
Kr-88	1.90×10^{-5}	1.06×10^{-6}	3.83×10^{-7}	2.13×10^{-8}	1.76×10^{-7}	9.83×10^{-9}	8.21×10^{-12}	3.86×10^{-14}
Kr-89	1.12×10^{-8}	6.35×10^{-10}	0.0	0.0	6.97×10^{-10}	3.96×10^{-11}	1.33×10^{-13}	6.42×10^{-16}
Xe-131m	6.17×10^{-5}	3.53×10^{-6}	5.94×10^{-5}	3.40×10^{-6}	6.62×10^{-9}	3.78×10^{-10}	2.72×10^{-13}	1.39×10^{-15}
Xe-133m	3.38×10^{-4}	4.57×10^{-6}	2.73×10^{-4}	3.70×10^{-6}	1.95×10^{-7}	2.63×10^{-9}	8.05×10^{-12}	9.08×10^{-15}
Xe-133	7.24×10^{-3}	4.47×10^{-4}	6.65×10^{-3}	4.10×10^{-4}	1.67×10^{-6}	1.07×10^{-7}	6.84×10^{-11}	3.75×10^{-13}
Xe-135m	5.84×10^{-7}	8.35×10^{-9}	8.76×10^{-10}	8.58×10^{-11}	2.61×10^{-8}	3.63×10^{-10}	2.50×10^{-12}	3.73×10^{-15}
Xe-135	6.17×10^{-5}	3.96×10^{-6}	1.82×10^{-5}	1.17×10^{-6}	1.97×10^{-7}	1.27×10^{-8}	8.41×10^{-12}	4.72×10^{-14}
Xe-137	2.18×10^{-8}	1.31×10^{-9}	0.0	0.0	1.33×10^{-9}	7.95×10^{-11}	2.35×10^{-13}	1.27×10^{-15}
Xe-138	3.32×10^{-7}	2.44×10^{-8}	1.36×10^{-27}	9.97×10^{-29}	1.51×10^{-8}	1.11×10^{-9}	1.44×10^{-12}	8.94×10^{-15}

The information presented in the above table was developed in support of the original license and is considered historical.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The Beaver Valley Power Station - Unit 2 (BVPS-2) design features comply with the intent of the guidance presented in 10 CFR Part 20 and Regulatory Guide 8.8 (USNRC 1978), Section C.2, regarding standards which limit radiation exposure to levels that are as low as reasonably achievable (ALARA). Compliance with other applicable regulatory guides and the general design criteria of 10 CFR Part 50, Appendix A, is discussed in Sections 1.8 and 3.1.2, respectively.

12.3.1.1 Implementation of ALARA

The design guidance of Section C.2 of Regulatory Guide 8.8, Revision 3, is implemented by means of an ALARA review utilizing a detailed checklist. This checklist provides a means of evaluating the plant design while ensuring that all points covered in Regulatory 8.8 have been addressed. The buildings, cubicles, and areas reviewed, include those adjacent to, and those having a potential for containing a radioactive source. Also included in the review are systems which have the potential for carrying radioactive fluids.

A detailed design review was conducted concurrently with detailed design of plant systems and layout. The review was performed throughout the design process to ensure that appropriate changes due to ALARA considerations are addressed and incorporated.

All findings, recommended actions, and resolutions are documented and a permanent ALARA review file is maintained and periodically reviewed to assure the resolution of outstanding items. Modifications occurring after completion of the detailed design review are subject to ALARA review of the modification itself and of any system it may impact.

12.3.1.2 Equipment Design Features

Plant arrangement features incorporate the following:

1. Equipment that may require servicing is designed and located to minimize service time. Radioactive equipment requiring servicing is located in individually shielded cubicles where practicable. The cubicles are designed such that the contribution from the principal radiation source in adjacent cubicles does not exceed the Zone IV dose rate levels given in Table 12.3-1. The actual dose rate in the cubicle is a function of the effectiveness of draining and flushing the equipment to be serviced and the number of penetrations and their manner of shielding. Zone II access is provided in passageways up to the cubicle entrance. Cubicle access opening size is dictated by potential

equipment removal and by the size of the maintenance equipment required. Pump cubicle openings for horizontal pumps are of sufficient size to skid the pump through the entranceway with the exception of the charging pump cubicles which have a moveable wall to allow for the removal or replacement of equipment using the auxiliary building crane. For other cubicles containing equipment requiring service, the openings are sized to allow the passage of the component.

Cubicle size is designed to allow sufficient clearance around equipment requiring service and to provide space for temporary shielding. The equipment service requirements for pull space and laydown space are provided within the cubicle without dismantling any piping other than that directly connected to the equipment. In certain cases where the equipment is always removed for dismantling, such as the heat exchangers, there is no laydown space within the cubicle, other than that required for the channel cover.

Layout in the cubicle also minimizes servicing time by placing the mechanical part of the equipment adjacent to the pipeway and the electrical portion toward the uninhabited access. Motor terminal boxes and other terminal boxes do not block access and are separated from the radioactive piping if possible. Platforms are provided where access is required for servicing. Stairs are provided, if possible, to these platforms.

2. Instruments requiring in situ calibration are located in the lowest practicable radiation fields.

3. Equipment and components requiring servicing are located in or designed to be moveable to the lowest practicable radiation fields.

4. The best available valves, valve packing, and gaskets are used to minimize leakage and spillage of radioactive materials.

   Valve selections are based on flow requirements, the best product available considering valve type, seat materials, and servicing conditions.

5. Penetrations of shielding and containment walls by ducts and other openings are designed to minimize exposure, and shield design specifications limit void content. Shielding material is used to plug openings between the wall and pipes, ducts, and sleeves where required.

   Piping penetrations through shield walls in the auxiliary building are either into adjacent cubicles, into the radioactive pipe chases for radioactive piping, or into the corridors for nonradioactive piping. Piping penetrations of the shield walls from differently zoned areas prevent line of sight from Zone II to any significant radiation source. This is accomplished by penetrating the walls at a location usually high and in the corner offset from the source. Most piping penetrations between cubicles are designed to prevent any line of sight from the significant radiation source to the adjacent cubicle. Exceptions are made to this criterion by the routing of the recirculation line from the liquid waste evaporator to the circulating pump and to the reboiler, as the interconnecting piping itself is a significant source. Another area where this criterion is not applied totally, is in the piping from the filter and demineralizer cubicle to the valve cubicles below. In this case, the valves are operated using extension stems to reach outside the cubicle. During maintenance of these valves, the source is reduced to ALARA by first flushing out the resin or removing the filter. The dose during maintenance is also limited to the hands of the personnel performing maintenance by providing sufficient space in the cubicle to work out of the direct line of sight and by providing portable shielding.

Electrical penetrations through shielded cubicles are located to preclude direct line of sight to any significant radiation source.

Instrument tubing penetrations through shielded cubicles are located to preclude direct line of sight to any significant radiation source.

Ventilation duct penetrations through shielded cubicles are made at the highest possible elevation and at a location to minimize direct line of sight. Any direct line of sight penetration is provided with shields located either inside or outside the cubicle penetration.

Shield walls are normally a part of the building structure and often serve as structural supports in addition to shielding. Specifications for concrete or other types of shielding include requirements for minimizing void content.

6. Radiation sources and occupied areas are separated if practicable; in particular, pipes or ducts containing potentially highly radioactive fluids will not pass through occupied areas unless adequately shielded.

   Radioactive piping (for example, process piping that carries radioactive materials) is located behind shielding to minimize radiation exposure to operating personnel. As required, pipe tunnels, chases, or shafts are properly segregated and shield nonradioactive piping, personnel passageways, and operating areas from radioactive piping. The shielding thicknesses are adequate to protect operating personnel from single or multiple pipe lines, depending upon the piping arrangement. All radioactive piping 2 1/2 inches and larger is specifically located on piping drawings and is accurately placed in the positions indicated on the drawings when the piping is installed in the field. Piping 2 inches and smaller is shown as field run on drawings, and for such piping, field personnel are responsible for preparing isometric drawings to reflect the relationship to both existing piping conditions and potentially radioactive systems. The drawings are reviewed prior to pipe installation which includes review to ensure that the piping runs are consistent with the ALARA concepts of the initial drawings. Regardless of size, radioactive piping is installed behind shielding, as needed, to minimize radiation exposure to operating personnel.

Physical barriers, such as doors, are provided for areas having radiation levels in excess of 100 mRem/hr. Locked barriers are provided for areas having radiation levels in excess of 1,000 mRem/hr.

7. Precautions are provided 1) to minimize the spread of contamination, and 2) to facilitate decontamination in the event of spillage.

   Tanks such as the liquid waste tanks, the evaporator bottoms tank, and the spent resin hold tank are placed in shielded cubicles. A 6-inch curb surrounds the tanks to contain minor spills. Drains within the cubicles are piped to the nearest auxiliary building sump. Other tanks, such as the steam generator blowdown test tanks and the boric acid tanks are also surrounded by curbs or dikes to contain spills or leaks and minimize the spread of contamination.

   Other vessels such as the demineralizer vessels and the filter vessels are in individual cubicles. The tank cubicles have their floor drains piped to the nearest auxiliary building sump.

   Other cubicles containing items such as pumps, heat exchangers, and valve stations are served by the vent and drain systems.

   The evaporator bodies are not provided with dike protection due to the necessity of locating the bottom cones through the floor and piping to the circulating pumps located below them. In this arrangement, a leak in the evaporator passes to the pump cubicle where it is piped through the floor drains to the sumps.

The concrete surfaces in the auxiliary building are painted in order to facilitate decontamination in the event of spillage. Water for flushing of walls and floors is available throughout the building from the fire control water system.

Contamination control areas are provided for the solid waste drumming area in the condensate polishing building and in the vicinity of the radiation protection access control area located at el 774 ft-6 in of the waste handling building.

A personnel decontamination area is located in the radiation protection area at the controlled access area checkpoint.

8. Interior surfaces as well as layout of ducts and pipes are designed to minimize buildup of contamination.

   Piping 2 1/2 inches or larger in diameter is butt-welded. This fabrication technique is also used on selected lines less than 2 1/2 inches in diameter, such as the spent resin transfer piping.

   Piping systems containing radioactive fluids are fabricated of stainless steel or other corrosion-resistant material.

   The piping systems are routed to avoid unnecessary sharp bends. Pockets and low points are also avoided.

   The valve stations in shielded cubicles are designed to minimize the buildup of crud in these areas by minimizing the number of pockets and stagnant vertical legs.

9. Systems that may become contaminated are designed to include provisions for flushing or remote chemical cleaning prior to servicing.

   The evaporator bottoms portion of the liquid waste system has permanently piped flushing connections. The evaporator is provided with a chemical cleaning connection.

10. The ventilation system is designed to ensure control of airborne contaminants, especially during maintenance operations when the normal air flow patterns may be disrupted (for example, open access portals).

    Airborne contamination is kept from spreading in the auxiliary building by the ventilation system. The building is continuously maintained under a slight negative pressure.

The auxiliary building ventilation system is designed to bring air from areas with lesser potential for contamination into areas with greater potential for contamination and to exhaust from the latter areas. A description of the auxiliary building ventilation system is provided in Section 9.4.3.

11. Wherever practicable, radiation and airborne contamination monitoring equipment with remote readout is included in areas to which personnel normally have access (where special conditions warrant, portable instrumentation may be substituted).

    Area and airborne radiation monitoring ensures that any substantial abnormal radioactivity release is promptly detected.

    An area radiation monitor is located to serve the waste evaporator area. Other areas also permanently monitored are the sample room and the doors leading to and from the auxiliary building to the condensate polishing building, and to the waste handling building. Normal personnel access to the auxiliary building, fuel building, waste handling building, and containment is through the monitor station in the radiation protection area.

    Airborne radiation monitors are provided for the containment, the auxiliary building regions, the waste handling building, and the fuel building (Section 12.3.4).

    The requirement for permanent monitors at other locations is not considered necessary due to the layout of the building which allows essentially shielded access to all radioactive equipment and piping areas. A description of the area and airborne radiation monitors is provided in Section 12.3.4.

12. The ventilation systems are designed for easy access and service to keep doses ALARA during alterations, maintenance, decontamination, and filter changes.

    The ventilation systems are arranged to provide the same shielded access to the charcoal and high efficiency particulate air filter trains as is provided for other radioactive equipment. Personnel protection features are discussed in Section 12.3.3.

13. Where practicable, shielding is provided between radiation sources and areas to which personnel may have normal or routine access, and shielding is designed for maintaining doses ALARA.

    Figures 12.3-1, 12.3-2, 12.3-3, 12.3-4, 12.3-5 and 12.3-6 show the shielding provided between radiation sources and areas to which personnel may have normal or routine access. Radiation zones are given in Table 12.3-1 with typical locations presented in Table 12.3-2.

    Radiation shielding is provided on the basis of maximum concentration of radioactive materials within each shield region. For batch processes, as an example, the highest radionuclide concentration in the batching process is assumed (just prior to draining of the tank).

    The bases for the selection of maximum radionuclide concentrations are described in Chapter 11. The sources in individual components are intentionally developed to simulate worst case conditions. The levels of activity under these conditions may thus be 10 times that of the expected levels in some components. This approach is carried out on a component-by-component basis so that shielding in each local area is adequate for the worst conditions (for example, activity in a tank just prior to drainage). Dose rates computed under such conditions ensure that the maximum allowable levels in each zone will not be exceeded under design conditions.

14. Moveable shielding and convenient means for its utilization will be available for use where permanent shielding is impractical.

    Steel or concrete supports for temporary shielding are provided as appropriate.

15. Adequate shielding is provided for radioactive wastes.

    Sources containing potentially high levels of radioactivity are completely shielded in the auxiliary, fuel, and waste handling buildings by full height walls.

    Solid waste is shielded by storage area walls in the solid waste area of the condensate polishing building.

    Process gas charcoal bed adsorbers and associated equipment are located in shielded cubicles.

16. Remote handling equipment is provided wherever it is needed and practicable.

Remote handling equipment is provided for removing filters from the filter vessels and for placing them into shipping containers. Contact operations in high dose rate areas are minimized.

The solid waste system is essentially a remotely operated system. Operations are conducted remotely or manually after the waste source has been shielded.

Stations for potentially radioactive valves are, in general, arranged either in shielded cubicles away from the equipment served and/or are provided with reach rods. The demineralizer and filter valves are in cubicles below and adjacent to the vessels and are also provided with reach rods.

17. All features for radiation control are designed to accommodate maximum expected failures such as fuel element cladding failures and steam generator tube leaks.

    Design features such as shielding and radiation zones accommodate clad defects for 1 percent failed fuel and primary to secondary steam generator tube leaks of 144 gal/day.

18. Sampling sites are located so exposures will be ALARA during such routine operations as sampling off-gas, primary coolant, and liquid waste.

    A sampling room is provided for the remote taking of routine samples from points in the reactor and auxiliary systems with the exception of the samples from the evaporators. Sample points for the evaporators are provided with a sample sink and ventilation hood, splash screen, and valves located outside the splash screen. The samples are provided with a recirculation path behind the shield wall at the sample sink with reach rods for the operator. A shielded sample station is also provided for solid waste streams (for example, evaporator bottoms, resins and sludges).

19. Redundancy of equipment is utilized in the plant design to facilitate ALARA by allowing longer holdup time between scheduled fluid processing and by reducing the urgency for accelerated equipment repair. Redundant radwaste solidification drum inspection/labeling stations permit processing more drums in shorter periods of time to reduce individual exposure to radioactive sources.

20. Materials

    Equipment specifications for components in the nuclear steam supply system contain specific limitations on the cobalt impurity content of the base metal, as given in Table 12.3-11, thereby controlling the potential for production of radioactive cobalt-60 from the base metal impurity cobalt-59. The estimated surface area of material in contact with the reactor coolant is given in Table 12.3-12. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Table 12.3-13 shows the estimated total surface area of stellite in the nuclear steam supply system. Nickel based alloys in the nuclear steam supply system (cobalt-58 is produced from activation of the base metal nickel-58) are similarly used only when component reliability may be compromised by the use of other materials. The major use of nickel based alloys in the nuclear steam supply system is the inconel steam generator tubes. The surface area in contact with the reactor coolant system is given in Table 12.3-12. From Tables 12.3-12 and 12.3-13, it can be seen that the inconel surface is the predominant area in contact with the reactor coolant system and that the stellite area is minimal. A further discussion of material considerations is given in Westinghouse (1977).

12.3.1.3 Design Objectives and Zoning Guidelines

The choice of facility equipment, layout and radiation shielding are designed in accordance with the requirements of 10 CFR 20 and Regulatory Guide 8.8. The zone guidelines and their bases described below are based on the provisions at 10 CFR 20.101 through 20.601 and 10 CFR 50 in effect during plant licensing and thus are part of the original design basis of the facility.

Zone I

The dose rate limit for Zone I is 0.75 mRem/hr, allowing continuous access. Based on 40 hr/wk occupancy, the Zone I dose rate limit assures that weekly doses will not exceed 1 percent of the maximum allowable quarterly whole body dose as specified in paragraph 20.101(b)(1) of 10 CFR 20. Facility and shielding are designed to maintain Zone I limits during full power operation with 1 percent failed fuel. Typical Zone I areas include the main control room, the outside surface of the containment structure, and all turbine plant and administration areas.

Zone II

The dose rate limit for Zone II is 2.5 mRem/hr, allowing periodic access. Based on 40 hr/wk and 50 wk/yr occupancy, the Zone II dose rate limit assures that the allowable whole body dose specified in paragraph 20.101(a) of 10 CFR 20 is not exceeded. During full power operation, with 1 percent failed fuel, facility and shielding design will maintain Zone II conditions for areas such as auxiliary and fuel building passageways and the containment structure personnel access hatch. During shutdown conditions, areas such as the containment structure operating floor and the containment structure outside of the crane wall will be Zone II.

Zone III

The dose rate limit for Zone III is 15 mRem/hr, allowing limited access. The Zone III limit provides approximately 6-hour access per week without exceeding the allowable whole body dose specified in paragraph 20.101(a) of 10 CFR 20. Valve operating stations are typical Zone III areas during full power operation with 1 percent failed fuel.

Zone IV

The dose rate limit for Zone IV is less than 100 mRem/hr. Such areas will require occasional access only. Typical Zone IV areas during full power operation with 1 percent failed fuel include the containment structure outside the crane wall at el 718 feet and above.

Zone V

The dose rate limit for Zone V is greater than 100 mRem/hr, requiring controlled access only. Such areas are provided with hardware to install locked doors for positive control of access in accordance with 10 CFR 20 and BVPS-2 Technical Specification 5.7. Typical Zone V areas include most auxiliary building cubicles and steam generator cubicles in the containment structure during full power operation with 1 percent failed fuel. During operation of the residual heat removal system (RHRS), all areas adjacent to the RHRS components and piping will become Zone V.

A summary of these zone designations is presented in Tables 12.3-1 and 12.3-2. Plant layout drawings with wall thickness and zone designations are presented on Figures 12.3-1, 12.3-2, 12.3-3, 12.3-4, 12.3-5, 12.3-6, 12.3-7, 12.3-8, 12.3-9, 12.3-10, 12.3-11 and 12.3-12. Radiation zones for shutdown and refueling are shown in Figures 12.3-13, 12.3-14, 12.3-15 and 12.3-16.

12.3.2 Shielding

In general, the biological shielding throughout the plant is designed to meet the specifications established in 10 CFR 20 and to keep personnel radiation exposure ALARA. More specifically, the criteria for penetrations, the zoning criteria, and the general guidelines stated in Subsection 12.3.1 are used as the basis for all shielding calculations and engineering decisions made in the plant arrangements.

Radiation shielding is designed to perform the following functions:

1. Limit dose rate levels to below the maximum dose rate assigned to the zone.

2. Maintain radiation doses to plant personnel within the limits established by 10 CFR 20.

3. Reduce radiation dose rates to personnel performing maintenance or inspection on plant equipment.

4. Limit the direct dose to personnel in the control room to less than 5 Rem whole body dose or equivalent under accident conditions as required by General Design Criterion (GDC) 19 of 10 CFR 50, Appendix A.

5. Reduce the direct dose rate at the exclusion area boundary under normal operating conditions and accident conditions.

Basic methods used for determining shield wall and slab thickness are described in the Topical Report RP-8A, Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants (Stone & Webster Engineering Corporation 1975).

Computer codes commonly employed for shielding analysis include the following:

1. GAMTRAN1 (SWEC 1982),
2. QADMOD (SWSC 1981a),
3. ANISND (SWEC 1981b), and
4. COHORT II (SWEC 1981c).
5. SW-QADCGGP (S&W 1999)

This combination of computer codes allows sufficient versatility to readily analyze any shielding problem. The bulk of shielding material is reinforced concrete with a minimum density of 135 lb/ft^3. Lead and steel are used where appropriate for the neutron shield tank support and around the fuel transfer tube. The supplementary neutron shield is made of borated, silicon-based (neutron attenuating) shielding material.

Shielding materials and thickness are listed in Table 12.3-3.

The original design basis for normal operation plant shielding is safe operation at a core power level of 2766 MWt, a one year fuel cycle length, and conservative reactor coolant source terms assuming 1% fuel defects. The design basis shielding design presented below in Sections 12.3.2.1 through 12.3.2.7, are based on the above design basis.

Power uprate represents a change from the original design basis. The assessment of impact of power uprate on adequacy of existing plant shielding was evaluated based on scaling techniques that took into consideration the radiation source terms used in the original plant shielding design for the reactor core and coolant (i.e., Tables 12.2-3 and 11.1-2, respectively) and the uprate reactor core and coolant source terms that are based on the analyzed core power level of 2918 MWt and with an 18-month fuel cycle length and are presented in Tables 15.0-7a and 15.0-8b, respectively.

Inside containment, due to the conservative analytical techniques used to establish original plant shielding design, the existing reactor primary and secondary shield including the fuel handling shields were determined to be adequate to address the approximately 6% increase in radiation source terms expected due to the uprate.

Shielding adequacy outside containment (where the radiation sources are either the reactor coolant itself, or down-stream sources originating from coolant activity), was determined by an evaluation that compared the uprate design primary coolant source terms (fission and activation products) to the original design basis primary coolant source terms. Three sources were considered: total primary coolant, degassed primary coolant and the primary coolant noble gas source. Due to the change in isotopic compositions and gamma energy spectrum between the original and the uprate reactor coolant fluid, the comparison was based on the dose rate resulting from the above sources shielded by 0, 1, 2, and 3 ft of concrete for representative source geometry.

The comparison showed that the ratio of the calculated dose rates resulting from the uprate source to the original design basis source, for the various design basis source term and shielding configurations discussed above, range from 0.82 to 1.97. However, since the design basis uprate primary coolant activity is a very conservative source term (i.e., based on 1% failed fuel, maximum core inventory with margin for instrument error, minimum letdown flow, etc.), credit was taken for a more realistic but limiting upper bound primary coolant activity based on the plant Technical Specifications.

The uprate assessment concluded that the Plant Technical Specifications will limit the uprate reactor coolant, degassed reactor coolant and reactor coolant gas activity and the associated dose rates assuming various shielding configurations, to less than or equal to the original design basis values.

Thus, taking into consideration the conservative analytical techniques used to establish the original shielding design, and the plant Technical Specifications which typically restrict the reactor coolant activity to levels significantly less than 1% fuel defects, the increase in the core power level and fuel cycle length is expected to have no significant impact on plant shielding adequacy and safe plant operation.

It is noted that operating personal at the station are protected by adequate shielding, monitoring, and operating procedures. Individual worker exposure is maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Procedural controls compensate, as necessary, for increased radiation levels to ensure that operator exposure remains ALARA, and that the normal operation radiation zones are labeled and controlled for access in accordance with the requirements of 10CFR20 related to allowable operator exposure and access control.

12.3.2.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel. The radiation consists of neutrons diffusing from the core, prompt fission gammas, fission product gammas, and gammas resulting from the slowing down and capture of neutrons.

The primary shielding is designed to:

1. Attenuate neutron flux to prevent excessive activation of components and structures.

2. Optimize the combination of primary and secondary shielding by reducing the radiation level from the reactor so that it is commensurate with the radiation levels from other sources.

The primary shield consists of a water-filled reactor vessel support structure having a radial dimension of approximately 3 feet surrounded by 4 1/2 feet of reinforced concrete. The reactor vessel support structure is designed to prevent overheating and dehydration of the concrete primary shield wall and to prevent activation of the station components within the containment structure. A thermosiphon cooling system is provided for cooling the water in the shield tank (Section 9.2.2.3).

The primary shield arrangement is shown on Figure 5.4-10.

A 17-foot high by 2-inch thick cylindrical lead shield is located beneath the reactor vessel support structure to protect personnel servicing the neutron detectors during reactor shutdown. A mat made of high temperature resistant, flexible insulation is draped underneath the reactor vessel to reduce activation of equipment and materials.

In order to maintain the shielding integrity of the primary shield, a supplementary neutron shield fabricated from borated silicon-based shielding material is provided around the reactor vessel and nozzles. The supplementary neutron shield is a collar/saddle design. The shield consists of 9.25 inches of shield material surrounding the microtherm insulation and air gap around the reactor vessel from the top of the support pads to the bottom of the seal plate. Additional shielding covers each nozzle radially from the vessel insulation to the primary shield wall. Above the nozzles are 1.0 ft^2 vent openings for normal venting and pressure relief. The collar extends vertically above the nozzle to the seal plate.

12.3.2.2 Secondary Shielding

Secondary shielding consists of reactor coolant loop shielding, reactor containment shielding, fuel handling shielding, auxiliary equipment shielding, and waste storage shielding.

Activation and fission products from the reactor coolant system (RCS) are the radioactive sources for which secondary shielding is required. This shielding is provided by a variety of structures and equipment.

12.3.2.3 Reactor Coolant Loop Shielding

Nitrogen-16 is a major source of radioactivity in the reactor coolant during normal operation and, therefore, establishes the basic shielding requirements for most of the internal containment cubicle walls. Activated corrosion and fission products in the RCS establish the shutdown radiation levels in the reactor coolant loop areas.

12.3.2.4 Containment Structure Shielding

The containment structure shielding consists of the steel inner-lined, reinforced concrete cylinder and hemispherical dome described in Sections 3.8.1 and 6.2. This shielding attenuates radiation during anticipated modes of reactor operation and design basis accidents. Radiation levels are reduced to design levels or below at the outside surface of the containment structure and at the site boundary.

Interior shield walls separate reactor coolant loop, steam generators, pressurizer, in-core instrumentation, and containment access sectors. This shielding allows access to the in-core instrument sector during normal operation and facilitates maintenance in all sectors during shutdown. Shield walls are provided around each steam generator above the operating floor to the height required for personnel protection.

12.3.2.5 Fuel Handling Shielding

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies and control rod assemblies from the reactor vessel to the spent fuel pool, while protecting personnel from the radiation emitted from these components.

The refueling cavity above the reactor is formed by a stainless steel-lined, reinforced concrete structure. This refueling cavity becomes a pool when filled with borated water to provide shielding during the refueling operation.

The cavity is large enough to provide storage space for the upper and lower internals and miscellaneous refueling tools.

The refueling cavity above the reactor vessel is flooded to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height ensures a minimum of 100 inches of water above a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate is less than 50 mRem/hr at the water surface.

During refueling, the dose rate at the walkway immediately adjacent to the refueling cavity is less than 15 mRem/hr when the raised spent fuel assembly is 8 feet below the surface of the water in the refueling cavity. Under the same conditions, the dose rate at the walkway immediately surrounding the fuel pool is less than 15 mRem/hr when the raised spent fuel assembly is approximately 8 feet from the edge of the reactor cavity.

Upon removal of the fuel from the reactor vessel, it is moved to the spent fuel pool by the fuel transfer mechanism via the refueling canal.

The shielding for the fuel transfer tube between the refueling cavity and the containment wall is shown on Figure 12.3-17. The shielding will be in place at all times except when repairs are being performed on the fuel transfer tube enclosure. The design limits dose rates during passage of a spent fuel assembly to approximately 50 mRem/hr on contact with the shield. Dose rates below the refueling cavity from the fuel transfer tube are also limited to approximately 50 mRem/hr on contact with the installed shielding.

Shielding for the fuel transfer tube between the containment and fuel building has been designed to prevent direct-line streaming through the seismic gap. The shield around the fuel transfer tube has a multiple labyrinth configuration with an equivalent thickness of 60 inches of concrete. Design details may be found on Figure 12.3-18.

Concrete shield walls are also provided for the fuel transfer canal. The wall thicknesses are given in Table 12.3-3.

The spent fuel pool in the fuel building is permanently flooded to provide a minimum of 100 inches of water above a fuel assembly being withdrawn from the fuel assembly transfer system. Based on the conservative assumption of a minimum decay time of 100 hours before fuel transfer and the minimum amount of water above the worst case spent fuel assembly, the dose rate at the surface of the fuel pool is less than 50 mRem/hr. The dose rate approximately 3 feet above the platform (8 ft above the surface of the pool) is less than 15 mRem/hr, which can be considered as the effective dose rate to operating personnel. Under the same conditions, the dose rate at the walkway immediately surrounding the fuel pool is less than 15 mRem/hr when the raised spent fuel assembly is approximately 8 feet from the edge of the fuel pool. Water height above stored fuel assemblies is a minimum of 23 feet. The sides of the spent fuel pool, three of which also form part of the fuel building exterior walls, are 6-foot thick concrete to ensure a dose rate of no more than 0.75 mRem/hr outside the building.

12.3.2.6 Auxiliary Equipment Shielding

The auxiliary components exhibit varying degrees of radioactive contamination due to the handling of various fluids. The function of the auxiliary shielding is to protect personnel working near the various auxiliary system components, such as those in the chemical and volume control system (CVCS), and the sampling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, decontaminate the entire system.

All ion exchangers and the most highly contaminated filters are located in separate cubicles adjacent to the southwest corner of the auxiliary building. Each ion exchanger or filter is enclosed in a separate, shielded compartment. The concrete thicknesses provided around the shielded compartments are sufficient to reduce the surrounding area dose rate to less than 2.5 mRem/hr. The shielding thicknesses around the mixed bed demineralizers are based upon a saturation activity which could produce a contact radiation level in excess of 10,000 Rem/hr.

In many areas, tornado missile protection, in the form of thick concrete, affords more shielding than that required for radiation protection.

12.3.2.7 Waste Storage Shielding

The waste storage and processing facilities in the auxiliary building, waste handling area, condensate polishing building, and decontamination facility are shielded to protect operating personnel in accordance with the radiation protection design bases set forth in Section 12.3.1.

Periodic surveys by radiation protection personnel using portable radiation detectors ensure that radiation levels outside the shield walls meet design specifications, and establish access limitations within the shielded cubicles. In addition, continuous surveillance is provided in selected areas by installed area radiation monitors. Area and airborne monitoring also ensures that any accidental radioactivity releases would be detected within a reasonable period of time. The largest accidental radioactivity release from the waste disposal system is analyzed in Section 15.7.1.

12.3.2.8 Accident Shielding

Accident shielding is provided by the containment structure, which is a reinforced concrete structure lined with steel. For structural reasons, the thicknesses of the cylindrical walls and dome are 54 inches and 30 inches, respectively. These thicknesses are more than adequate to limit the direct radiation dose at the exclusion boundary to acceptable values. Shielding is also provided for the recirculation spray system cubicles, hydrogen control system cubicles, and post-accident sampling system.

12.3.2.9 Control Room Shielding

Shielding provided by the main control room walls, and the separation of the main control room from the containment structure, ensures that an operator would be able to remain in the main control room for 30 days after an accident and not receive an integrated dose in excess of 5 Rem (TEDE). This dose includes the 30 day direct radiation dose from activity inside the containment, the external radiation contribution from the postulated radioactive plume resulting from containment leakage, engineered safety features (ESF) system leakage, and RWST back-leakage. Leakage from the containment is assumed for 30 days following the accident, as discussed in Chapter 15.

The radiation sources in the containment during the assumed DBA are calculated by the methods described in Section 15.6.5. These sources are assumed to be evenly distributed throughout the containment. The containment is then treated as a volume source and the 30 day direct radiation dose inside the 24-inch thick concrete wall of the main control room is calculated using typical shielding computational techniques.

The released activity is assumed to leak from the containment structure at a rate of 0.1 percent of the contained volume per day for the first 24 hours, and at half that rate thereafter. The plume is converted into a semi-infinite volume source surrounding the main control room.

Sources such as containment, ESF system leakage, and RWST back-leakage that contribute to personnel doses from the intake of the outside atmosphere into the control room, are described in Section 6.4.2.5.

The integrated dose from all sources described in Section 6.4.2.5 is provided in Table 15.0-13. This dose is well below the dose criterion of 5 Rem TEDE. Thus, the main control room walls, which must be a minimum of 24 inches thick for tornado missile protection, provide more than adequate shielding from radiation.

Special consideration has been given to the design of penetrations and structural details of the main control room so as to establish an acceptable condition of leaktightness.

The control room air-conditioning systems are installed within the spaces served and are designed to provide uninterrupted service under accident conditions. Upon a containment isolation Phase B (CIB) signal or a high radiation signal from the control room redundant area monitors, the normal replenishment air and exhaust systems are isolated automatically from the main control room by tight closures in the ductwork.

The control room emergency ventilation system is actuated to pressurize the control room envelope. The main control computer, and mechanical rooms are included in the control room envelope.

The radiation levels in the main control room are measured by installed area and airborne monitors to verify safe operating conditions.

12.3.2.10 Radiation and Shield Design Review

A radiation and shield design review was performed in accordance with NUREG-0737, Action Item II.B.2 (USNRC 1980), in order to ensure personnel accessibility after a design basis accident (DBA). The DBA considered for this evaluation was the loss-of-coolant accident (LOCA). Source terms for the LOCA are discussed in Section 12.2.1.3.

Table 12.3-4 identifies areas considered vital due to occupancy requirements or by virtue of being an access route to a vital area following a DBA. Additional information provided in this table includes an estimate of the anticipated times after a LOCA that access is required, and a conservative estimate for the stay times needed in the vital areas. This table also presents calculated dose values for the primary and applicable alternate access routes. The post-accident access routes considered are identified and controlled administratively by procedure.

The following general assumptions and criteria are used as a basis for review of all vital areas and access routes as applicable:

1. The starting point for all activities is the Beaver Valley Power Station - Unit 1 (BVPS-1) locker room located west of the BVPS-1 control room.

2. Access areas may have two access paths defined; the one denoted as the primary route is chosen on the basis of lower potential exposure, and the second denoted is the alternate route. Access areas can have one path defined because the significant dose contributions for these routes are due to travel outside the buildings, thereby minimizing problems with pathway obstructions and localized high radiation areas. Access areas 1 and 2 have no access paths associated with them, hence have no alternate path.

3. Each of the vital access areas was evaluated using conservative assumptions, and the doses were determined to be less than the dose limits of GDC 19 following a DBA using either the primary or alternate routes. Vital access areas 1 and 2 (the control room and ERF) have been evaluated for habitability for 30 days following a LOCA, assuming alternative source term methodology and using the associated 10 CFR 50.67 dose limit of 5 Rem TEDE.

4. Personnel travel is based on:

    a. A constant walking speed of 3 ft/sec (approximately 2 mph) for travel within the plant or,

    b. A constant driving speed of 30 ft/sec (approximately 20 mph), for travel between BVPS-2 and the ERF.

5. Operator exposure due to airborne activity from containment leakage, or leakage from systems located outside the containment is not considered for the access study. However, these sources are considered for the control room and the ERF habitability studies. Direct shine from ESF ventilation filters is also included in the vital access studies.

6. Areas requiring continuous occupancy were evaluated to ensure that the dose rate averaged over 30 days would be less than 15 mRem/hr.

The following areas and systems specifically identified in NUREG-0737 (USNRC 1980) are addressed under other names, or do not apply to the BVPS-2 design:

1. Motor control centers - completely automatic with no manual functions required.

2. Containment isolation reset control area - located in the control room.

3. Emergency power supplies - addressed in Accesses 6, 10, 11, and 12.

4. Instrument panels - the only panels requiring access are defined in Access 4 (hydrogen analyzer panels).

5. Manual emergency core cooling system alignment area - addressed in Access 9.

6. Radwaste control panels - BVPS-2 is not designed to use the liquid waste, gaseous waste, or solid waste systems for post-accident cleanup.

7. Residual heat removal system - located in the containment and not considered for post-accident access evaluation.

8. Chemical and volume control system (CVCS) - the system is immediately isolated from the primary coolant system by the letdown line isolation valve due to a Containment Isolation Phase A (CIA) signal and is not considered to be a source for post-accident access. The charging pumps and associated piping are used as part of the high head safety injection system. Therefore, this part of the CVCS is considered as a post-accident source.

9. Gaseous radwaste system - isolation of the CVCS from the post-accident source will also isolate the gaseous waste system.

10. License Amendment 123 eliminated the requirement to have and maintain a Post Accident Sampling System (PASS), and utilize this system in the short term following an accident. Therefore, the system is not considered vital and the associated design review information has been removed from the UFSAR.

11. License Amendment 142 eliminated the requirement for hydrogen control using the hydrogen recombiner. Consequently, the associated design review information has been removed from the UFSAR.

A description of the post-accident activities required to ensure that vital areas are accessible is provided as follows for each area described. Access routes 13 and 14 expose an operator to direct shine from the SLCRS filters, which could become increasingly radioactive due to accumulation of containment or ECCS leakage on the filter media over 30 days. However, the SLCRS filters are located inside a shielded cubicle, and the photon energies from sources on the filter media are relatively weak. As a result, vital areas served by routes 13 and 14 have been shown to remain accessible over the 30-day duration of the accident.

The impact of core power uprate to an analyzed power level of 2918 MWt and use of an 18 month fuel cycle on the post-LOCA gamma radiation environmental dose rates at BVPS-2 was evaluated based on a comparison of the gamma source terms developed based on the original core inventory used to develop the post-LOCA dose rates, to the uprated gamma source terms. This scaling approach was utilized to estimate the operator exposure during Access 3 through 14.

For the unshielded case, the factor impact on post-accident gamma dose rates was estimated by ratioing the gamma energy release rates weighted by the flux to dose rate conversion factor, as a function of time, at the uprated power level, to the corresponding weighted source terms based on the original analysis.

To evaluate the factor impact of the uprate on post-LOCA gamma dose rates (vs. time) in areas that are shielded, the original and uprated source terms discussed above were weighted by the concrete shielding factors for each energy group. The concrete shielding factors, for 2 feet of concrete (representative of moderate shielding), and 4 feet of concrete (representative of heavy shielding), provide a basis for comparison of the post-LOCA spectrum hardness of source terms with respect to time for both original design and the uprate case.

Since the gamma dose rate scaling factors will vary with source, time as well as shielding, to cover all types of analysis models/assessments, the maximum dose rate scaling factor developed from all of the above assessments was used, for the most part, for all source/receptor combinations, with or without shields, and at all time periods after LOCA. Exceptions included a few cases where analysis refinement was performed to support operator exposure compliance within regulatory limits at uprate conditions.

Access 1 - Control Room

The BVPS-2 control room will be continuously manned throughout the DBA. The control room shielding and ventilation system design ensure habitability for 30 days while staying within the dose limits of 10 CFR 50.67. Details are presented in Section 12.3.2.9.

Access 2 - Emergency Response Facility

Continuous occupancy of the ERF is required following a DBA. Shielding and filtered ventilation system designs are provided but have been shown to be not required in order to ensure occupancy of the ERF for 30 days while staying within the dose requirements of 10 CFR 50.67.

Access 4 - Operation of Post-accident Hydrogen Analyzers

The hydrogen analyzers are started automatically upon the initiation of safety injection (CIA signal). However, for this evaluation, it is assumed that access is required to the hydrogen analyzer control panels located at el 730 ft-6 in of the service building within 30 minutes of the start of the DBA. The entry is made to ensure proper operation of the hydrogen analyzers and to take local readings. The task is assumed to require approximately 15 minutes stay time in the vicinity of the control panel after which time the operator returns to the BVPS-1 locker room.

Specific details about the hydrogen control system are presented in Section 6.2.5.

Access 5 - Obtaining and Analyzing Post-Accident Samples

As discussed above, the PASS is not considered vital and the PASS shield design review information has been removed from the UFSAR.

Access 6 - Lubrication Oil for the Emergency Diesel Generators

Following an accident and subsequent emergency start-up and extended operation of an emergency diesel generator access to the diesel generator building would be required to add lubrication oil to the engine lubrication oil sump.

At the worst case oil consumption rate, access would be necessary approximately 100 hours after start-up of the diesel generator. Three 55-gallon drums of lubrication oil would have to be transported from the oil storage room of the south office and shop building to the diesel generator building and emptied into the lubrication oil sump.

The analysis assumed access to the diesel generator being necessary at 96 hours post-accident. The time required to transport the three oil drums and to transfer the oil to the oil sump was assumed to be 3 hours.

Access 7 - Operation of the Hydrogen Control System

As discussed above, operation of the hydrogen control system (HCS) is no longer required and the associated design review information has been removed from the UFSAR.

Access 8 - Personnel Movement

Integrated doses resulting from personnel movements to provide support for vital operations at BVPS-2 and operator shift changes in the control room following a DBA versus time after the accident are presented in Table 12.3-7. The dose values represent one way trips from the ERF to the BVPS-2 control room/BVPS-l locker room or from the BVPS-2 control room/BVPS-l locker room to the ERF at selected times from the start of the DBA to 30 days post-accident. The dose is due to radiation exposure for approximately 3 minutes of walking and 1 minute in a transport vehicle.

Access 9 - Manual Operation of Valves

In the unlikely event of a valve failure, certain systems were designed so that critical valves can be operated via a remote mechanical action. A listing of valves which are critical to the safe shutdown of BVPS-2 is given in Table 12.3-8 for the following systems:

1. Service water system (SWS),

2. Recirculation spray system,

3. Quench spray system,

4. Safety injection system.

All manual valve operation considered for this design take place at el 741 feet of the safeguards building. Because all systems considered are 100 percent redundant, it is assumed that manual operation of valves will not be required for the first 24 hours after the DBA. The estimated stay time for the access route is 15 minutes. Following the manual operation of valves, the operator will return to the BVPS-1 locker room.

Access 10 - Residual Heat Removal Suction Valve Transfer

Residual heat removal (RHR) motor operated suction valves are designed so that the power supply to their motors can be manually transferred. In order to accomplish the transfer, electrical equipment located at el 738 ft - 10 in. of the cable vault area must be manually operated. Because the only anticipated effect of a failure in the RHR is an extension of cool down time, it is conservatively assumed that access to the electrical equipment will not be required for the first 24 hours after the DBA.

The estimated stay time in the area necessary to complete the manual transfer is 15 minutes, after which the operator will return to the BVPS-1 locker room. Specific details about the RHR system can be found in Section 5.4.7.

Access 11 - Electrical Connection of Permanently Installed Spare Equipment

In the unlikely event of a failure, certain systems include in their design some spare critical components. All spare equipment must be connected manually to an electrical power supply. Systems considered and described in their respective sections include:

1. Charging system - Section 9.3.4,

2. Service water system - Section 9.2.1, and

3. Component cooling water system - Section 9.2.2.

Spare equipment which will require manual connection includes:

1. Charging system pump 2CHS*P21C(SG),

2. Service water system pump 2SWS*P21C(SG), and

3. Component cooling water system pump 2CCP*P21C(SG).

Because all systems considered for this design review are 100-percent redundant, it is conservatively assumed that manual connection of any spare equipment will not be required for the first 24 hours after the DBA. Estimated stay time in the area is 15 minutes, after which the operator returns to the BVPS-1 locker room. The spare charging pump, service water pump, or component cooling water pump may be manually connected at el 730 ft-6 in of the service building.

Access 12 - Ventilation Fan for the Service Water System

There are three separate, independent, and redundant intake pump cubicles utilized for BVPS-2, each with its own cubicle housing a ventilation system. Thus, the failure of one system will not interfere with the operation of the others.

In the unlikely event of a failure of one of the ventilation fans for the SWS, the swing ventilation fan may be connected manually to either emergency power supply, along with the swing service water pump. It is conservatively assumed that access is not required for the first 24 hours after a DBA.

Manual connection of the swing ventilation fan and service water pump takes place at el 705 feet of the main intake structure. The time spent connecting the swing fan and pump is assumed to be 15 minutes each for a total stay-time in this area of 30 minutes after which the operator returns to the BVPS-1 locker room.

Access 13 - Obtaining and Analyzing a Post-Accident Effluent Sample

A sample of the post-accident gaseous effluent must be obtained and analyzed within 3 hours post-accident. The sample is obtained at the elevated release monitor (ERM) skid on el 773 ft-6 in. of the auxiliary building and analyzed at either the primary chemistry laboratory on el 774 ft - 6 in. of the condensate polishing building or at the ERF.

Access to the ERM is assumed to be at 1 hour post-accident. The stay time at the ERM to obtain the sample is estimated to be 10 minutes. It is assumed that the sample chamber is shielded such that the operator will receive a negligible dose from the sample itself both during change out and during transportation. The sample will be taken for analysis to either the primary chemistry laboratory counting room or by vehicle to the ERF. A stay time of 1 hour is assumed necessary to perform the analysis. If the sample is taken to the counting room, after performing the analyses the operator will return to the BVPS-1 locker room. In the event the sample is taken to the ERF, a vehicle will be used to transport both the sample and operator. It is assumed that the operator will not return to the BVPS-1 locker room from the ERF.

Access 14 - Re-energizing ECCS Valve Circuits

Access to the motor control centers, MCC*2E03 and MCC*2E04, on el 755 ft-6 in. of the auxiliary building is required to complete the switchover from cold leg injection to the containment sump recirculation mode. Access is needed to re-energize motor-operated valve circuits by closing the associated breakers.

Access is assumed to occur at 25 minutes post-accident with a stay time of 5 minutes at the MCC location. An operator will travel from the BVPS-1 locker room to the MCC location in the auxiliary building, close the breakers, and then return to the BVPS-1 locker room.

Summary

Areas requiring access due to vital requirements have been defined, evaluated, and the results summarized in Table 12.3-4. The vital access study indicates that the plant shielding provides adequate protection to operators to ensure compliance with the GDC 19 dose requirements.

12.3.3 Ventilation

12.3.3.1 Design Objectives

The function and design bases of the ventilation systems are given in Section 9.4. Consistent with these bases, the following specific objectives pertain to radiation protection and the commitment that occupational radiation exposures will be ALARA in accordance with the intent of Regulatory Guide 8.8.

1. The airborne radioactivity inside BVPS-2 buildings, other than the containment structure, during normal operation and under anticipated operational occurrences is less than the concentrations given in Column 3, Table I of Appendix B to 10 CFR 20. Each system has adequate capability to reduce concentrations of airborne radioactivity in areas not normally occupied, where maintenance or in-service inspection has to be performed, to levels in accordance with the requirements of 10 CFR 20.103.

2. Concentrations in areas accessible to administrative personnel are less than the concentrations given in Column 1, Table 2 of Appendix B to 10 CFR 20.

3. The airborne concentrations in all plant areas are ALARA.

4. Deleted

5. The containment purge air system is capable of reducing airborne radiation levels in the containment to acceptable levels prior to and during extended personnel occupancy of the containment.

6. Air flow within the auxiliary, waste handling, and fuel buildings is normally from areas of lower to higher potential airborne contamination and then to monitored vents.

7. Systems are designed so that filters containing radioactivity can easily be maintained to minimize the radiation dose to maintenance personnel.

The design and expected airborne radioactivity levels during normal operation and under anticipated operational occurrences for plant buildings are given in Section 12.2.2. The expected inhalation dose rates to plant personnel are given for each building in Section 12.4.2.

12.3.3.2 Design Description

Detailed description of the ventilation systems for those plant buildings which contain radioactivity or potentially radioactive systems are given in the following sections:

Section	Title
9.4.1	Control Building Ventilation System
9.4.2	Spent Fuel Pool Area Ventilation System
9.4.3	Auxiliary Building and Radwaste Area Ventilation System
9.4.4	Turbine Building Area Ventilation System
9.4.5	Engineered Safety Features Ventilation Systems
9.4.7	Containment Ventilation System
9.4.9	Main Steam and Feedwater Valve Area Ventilation System

9.4.11 Safeguards Area Ventilation System

9.4.12 Cable Vault and Rod Control Area Ventilation System

9.4.13 Decontamination Building Ventilation System

9.4.15 Gland Seal Steam Exhaust Ventilation System

9.4.16 Condensate Polishing Building Ventilation System

12.3.3.3 Personnel Protection Features

The recommendations of Regulatory Guide 1.52 as described in Section 1.8 are implemented in the design of the safety-related ventilation filter trains to help ensure that occupation radiation exposures will be ALARA.

This is accomplished by utilizing the following criteria:

1. Each filter train is housed in a shielded compartment, room, or cubicle except for the control building filters which occupy a common cubicle with the air conditioning unit.

2. Adequate aisle space is provided for both personnel and equipment adjacent to the service side of the filter trains, and above those sections which require top access (charcoal adsorber). Adequate space is also provided on the other three sides.

3. Convenient and accessible passageways and corridors from the filter trains to the elevators and equipment hatches are provided for transport of replaceable filter train components and the equipment used in accomplishing their replacement.

4. Replaceable elements are designed for ready removal from the clean filter side, and minimal radiation exposure of personnel. A portable vacuum conveying system will be provided for removing and replacing gasketless-type charcoal adsorbers. Contaminated filters can be transported in shielded containers if necessary.

5. Rigid, hinged access doors are provided in accordance with ANSI N-509 for man-entry filter trains.

6. High-efficiency particulate air filters and prefilter arrangements facilitate easy replacement.

7. The proper distance for ease of maintenance and removal of filter elements will be maintained from mounting frame to mounting frame between banks of components.

8. Adequate, vapor-tight lighting will be provided on each side of the filter banks for man-entry filter trains.

9. Filter support and alignment cradles are provided for aligning and supporting filter elements during filter change, except for side replaceable filters.

10. Drains are provided to convey water from moisture separators, maintenance, or fire protection discharge, out of the filter train.

11. Permanent test fittings are provided for initial and periodic field testing.

A sample layout of an air cleaning system housing is shown on Figure 12.3-32. The filter housings are mounted on concrete slabs with sufficient space available to allow unimpeded access to the filter assemblies. Isolation is achieved through motor-operated dampers (Figure 6.5-2). Requirements for testing are met with test canisters and parts provided with the filter assemblies. Detailed operating procedures are available for the decontamination process, should the need arise.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Design Criteria

The monitoring instrumentation shall meet the following criteria:

1. Airborne monitors shall provide operator information relative to airborne concentrations of radioactive gases and particulate radioactivity at various locations in the plant.

2. Area monitors shall provide operator information relative to external gamma radiation levels at fixed points throughout the plant.

3. The monitors shall be responsive to dose rates that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for anticipated operational occurrences.

4. Category I radiation monitoring instrumentation shall have available power from Class 1E buses in the event of a power failure or postulated accidents. Category II radiation monitoring instrumentation shall be powered from non-Class 1E power, which is a highly reliable power source, and is backed up by the nonsafety-related onsite diesel generator.

5. The monitors shall be calibrated routinely and after any maintenance work is performed on the instrument.

6. Each monitor shall have local audible and visual alarms, variable alarm set points, and a local readout device.

7. Each monitor shall have provisions in the control room for readout and alarms to indicate equipment malfunction, alert radiation level, and high radiation level.

8. Recording capability is provided for all monitors.

Each monitor, excluding the in-containment high-range area monitors, has a radiation detector with a remotely operated check source. Each detector has associated electronic equipment and cabling.

The continuous airborne radioactivity monitoring system meets the following criteria in addition to that listed previously.

1. Capability to sample air at selected locations where airborne radioactivity is most likely to exist, such as waste handling areas, the fuel building and reactor operating floor. Air being exhausted during normal operation is monitored by instruments capable of detecting ten MPC-hour (particulate or gaseous radioactivity) in any compartment which has a possibility of containing airborne radioactivity and which also may be occupied by personnel.

2. Capability to measure representative air concentrations at the detectors located as close to the sampler intakes as practicable.

3. Provisions for sampling of ventilation systems upstream of HEPA filters.

These systems are provided in accordance with Regulatory Guide 8.2, Guide for Administrative Practices in Radiation Monitoring (USAEC 1973), with respect to providing instrumentation for evaluation of potential radiation hazards. In addition, the following guidance is employed: American National Standard ANSI N13.1-1969, for guidance on sampling airborne radioactive materials in nuclear facilities and Regulatory Guide 8.8, for guidance on monitoring systems. The locations of the area and airborne monitors are shown on Figures 12.3-2 through 12.3-6.

The digital radiation monitoring system includes the following radiation monitors, grouped according to their classification as airborne or area monitors (quantity in parentheses):

1. Airborne Radiation Monitors

    a. QA Category I Seismically Qualified

Containment Airborne (1)

b. QA Category II Seismically Qualified

1) Fuel building ventilation (1)

2) Leak collection vent (1)

3) Auxiliary building 718-A (1)

4) Auxiliary building 718-B (1)

5) Auxiliary building 718-C (1)

6) Auxiliary building 735-A (1)

7) Auxiliary building 735-B (1)

8) Auxiliary building 755-A (1)

9) Auxiliary building 755-B (1)

10) Waste handling building (1)

11) Control room airborne (1)

12) Condensate polishing building (1)

2. Area Radiation Monitors

a. QA Category I Seismically Qualified

1) In-containment high-range area (2) - post-accident

2) Control room area (BVPS-2) (2)

3) Outside personnel hatch area monitor (1)

b. QA Category II Seismically Qualified

1) Safeguards recombiner area (2)

2) Sample room area (1)

3) Manipulator crane area (1)

4) Incore instrumentation equipment area (1)

5) Decontamination area (1)

6) New fuel storage area (1)

7) Fuel pit bridge area (1)

8) Auxiliary building area (9)

9) Waste handling area (4)

10) Reactor containment area low range (1)

11) Condensate polishing area (5)

12) Spare area (3)

13) Primary access facility (1)

12.3.4.2 Airborne Radioactivity Monitoring

Fixed airborne radiation monitoring instruments are located in selected areas throughout BVPS-2. The location and number of these instruments ensure a continuous flow of information to operating personnel concerning the airborne radioactivity levels in selected plant areas and in ventilation effluent streams. The instruments are designed for service based on expected radioactivity levels during normal operation and anticipated occurrences.

Airborne radioactivity detection is accomplished by a sampling/sensing element configuration integral in the airborne radiation monitors. The airborne radiation monitors, in conjunction with the process and effluent radiation monitors and area radiation monitors, constitute the digital radiation monitoring system (DRMS). The DRMS is a digital microprocessor-based system described in Section 11.5. Airborne radiation monitors are off-line sampler types equipped with particulate and gaseous detectors and a readily removable charcoal filter cartridge. The charcoal filter cartridges may be taken to the radiation protection laboratory and analyzed to determine the quantity of deposited iodine from which airborne concentrations may be derived.

Sensitivities and ranges for these monitors are given in Tables 12.3-9 and 12.3-10. ANSI N13.10 will be used as a guide for establishing sensitivities of these monitors.

Containment Atmosphere Radioactivity Monitor

The containment atmosphere radioactivity monitor draws a sample from the containment atmosphere recirculation system, Section 9.4.7, and monitors the radioactivity concentration in the containment structure. This monitor is located in the rod control and cable vault area, el 738 feet with the sample line run as short as possible with a minimum of bends. The containment airborne monitor may be employed to provide a sample of the containment atmosphere for laboratory analysis. In the event of a LOCA, a CIA signal closes the containment isolation valves in the containment atmosphere monitoring sample lines.

The containment atmosphere monitor system monitors both particulate and gaseous activity and is used to help detect RCS leakage. The particulate activity monitor has a measurement range of 10E-10 to 10E-5 μCi/cc. The gaseous activity monitor has a measurement range of 10E-6 to 10E-1 μCi/cc.

This monitor is seismically qualified and supplied with power from the uninterruptible ac electric power system.

<u>Waste Handling Building Airborne Monitor</u>

The waste handling airborne monitor draws a sample from the waste handling building ventilation and monitors the radioactivity concentration in the waste handling building. The monitor is located at el 773 feet of the auxiliary building.

<u>Auxiliary Building Airborne Monitors</u>

The radioactivity concentration in the auxiliary building is monitored by seven airborne radiation monitors. Each monitor draws a sample from specific locations in the flow path of the auxiliary building ventilation system ductwork.

The sample points are located such that each monitor will determine the airborne radioactivity concentration in a portion of the auxiliary building. This arrangement will facilitate rapid determination of the location of any source of high airborne radioactivity.

These monitors are located at el 735 feet and el 755 feet of the auxiliary building.

<u>Condensate Polishing Building Airborne Monitor</u>

The condensate polishing building airborne radiation monitor draws a sample from the building's normal ventilation system and monitors the radioactivity concentration in the condensate polishing building. This monitor is located at el 794 feet of the condensate polishing building.

<u>Control Room Airborne Monitor</u>

The control room airborne radiation monitor draws a sample from the control room air ventilation system and monitors the radioactivity in the control room. This monitor is located at el 735 feet of the control building.

Fuel Building Airborne Monitor

The fuel building airborne radiation monitor draws a sample from the fuel building ventilation exhaust and monitors the radioactivity concentration in the fuel building. The monitor is located at el 766 feet of the fuel building.

Leak Collection Airborne Monitor

The leak collection airborne radiation monitor draws a sample from the leak collection system exhaust duct and monitors the radioactivity concentration in the contiguous areas. The monitor is located at el 773 feet of the auxiliary building.

12.3.4.3 Area Monitoring

The area radiation monitors provide continuous surveillance of radiation levels in selected areas throughout BVPS-2. These selected areas include locations where personnel may be present and where significant radiation levels may occur.

The area radiation monitors are part of the DRMS described in Section 11.5. This system is based on distributed microprocessors and redundant central processor units.

The microprocessor will provide controls, alarms, indication, and data processing as described in Section 11.5.

The alarm set-point for each monitor is at a level to provide sufficient warning of high radiation levels to operating personnel. Alarm set-points are based on design dose rates for the different zones as outlined in Section 12.3.1.

The operability of each detector is checked, as required, with the check source which may be actuated remotely.

All area radiation monitors are equipped with Geiger-Mueller detectors, except the in-containment high range monitors and the high range detector associated with the outside personnel hatch monitor, which are equipped with ionization chambers.

The area monitor, which is located outside the containment opposite the personnel hatch, will monitor high gamma dose rates from the containment structure in the event of a design basis accident. This monitor will serve as a backup to the two in-containment high range area monitors.

The redundant, safety-related area monitors centrally located in the BVPS-2 control room will isolate the control room envelope on a high radiation signal. This alternate method for control room isolation (further described in Sections 6.4 and 9.4.1) protects the operator against the radiological consequences of DBAs that do not initiate a CIB signal.

Two in-containment high-range monitors, capable of detecting and measuring the radiation level within the reactor containment during and following an accident, are provided in accordance with Regulatory Guide 1.97 and Item II.F.1, Attachment 3 of NUREG-0737 (USNRC 1980). The monitors meet the requirements of Table II.F.1-3 of NUREG-0737 (USNRC 1980) and Regulatory Guide 1.97.

Two safeguards recombiner area monitors are provided outside the cubicles in which the post-DBA hydrogen recombiners are located.

A list of area radiation monitors and their locations, sensitivities, and ranges is presented in Table 12.3-10. The sensitivity of each monitor is the lower value of the monitored range.

12.3.4.4 Criticality Monitoring

10 CFR 70.24, "Criticality Accident Requirements" requires that licensees install criticality monitors in areas where large quantities of Special Nuclear Material are handled or stored. An exemption to 10 CFR 70.24 was obtained for BVPS Unit 2 such that a criticality monitoring system was not required.

Subsequently, 10 CFR 50.68, "Criticality Accident Requirements" was published as an alternative to 10 CFR 70.24. 10 CFR 50.68 lists specific requirements that need to be met in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24. BVPS Unit 2 has chosen to comply with 10 CFR 50.68 as a means of eliminating the requirement for a criticality monitoring system.

12.3.5 Testing, Inspection, Calibration, and Maintenance

Preliminary testing is performed as described in Section 14.2.12. In-service inspection, calibration, and maintenance is addressed in Section 11.5 and will be performed, as necessary, in accordance with the BVPS-2 Technical Specifications, the Licensing Requirements Manual, and normal station maintenance procedures.

12.3.6 References for Section 12.3

Stone & Webster Engineering Corporation (SWEC) 1975. Radiation Shielding Design and Analysis Approach for Light Water Reactor Plants. Topical Report RP-8A, Boston, Mass.

SWEC 1981a. QADMOD - Point Kernel Gamma Transport. NU-137.

SWEC 1981b. ANISND - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering. NU-146.

SWEC 1981c. COHORT2 - Monte Carlo Radiation Environment Analysis. NU-157.

SWEC 1982. GAMTRANI - Gamma Transport by Point Kernel Technique. NU-003.

U.S. Nuclear Regulatory Commission 1980. Clarification of TMI Action Plan Requirements. NUREG-0737.

Westinghouse 1977. "Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures As Low As Reasonably Achievable," WCAP-8872, April 1977.

Tables for Section 12.3

TABLE 12.3-1

RADIATION ZONE DESIGNATIONS

Zone No.	Zone Description	Maximum Dose Rate (mRem/hr)
I	Unrestricted area-continuous access	< 0.75
II	Restricted area-periodic access	< 2.5
III	Restricted area-limited access	< 15
IV	Radiation area-infrequent access	< 100
V	High radiation area-not normally accessible (controlled access provided)	≥ 100

TABLE 12.3-2

TYPICAL LOCATIONS FOR DESIGNATED RADIATION ZONES

Zone No.	Typical Locations
I	Control room and all administrative areas
II	Auxiliary, waste handling, condensate polishing, and fuel building passageways in general
III	Valve operating stations in auxiliary, waste handling system and fuel buildings
IV	The containment area outside the crane wall
V	The containment area inside the crane wall, radioactive equipment cubicles, and solid waste storage area

TABLE 12.3-3

CONTAINMENT AND CONTAINMENT CONTIGUOUS AREAS SHIELDING SUMMARY

Shield Description	Material	Thickness (in)
Reactor vessel support structure		
Above el 705' 8"	Water	34
	Steel	3
Below el 705' 8"	Steel	1 1/2
	Lead (+1/2") steel cladding)	2
Primary shield	Concrete*	54
Supplementary neutron shield	Borated, silicon-based shield	9 1/4
Cubicle - crane support wall	Concrete	24-33
Crane support wall above operating floor	Concrete	24
Containment wall	Concrete	54
Containment dome	Concrete	30
Floor-el 718' 6"	Concrete	24 and 54
Operating floor	Concrete	24 and 47
Refueling cavity wall	Concrete	42
Missile shield above CRDM housing	Concrete	21
Refueling cavity water (minimum depth of water above fuel being transferred)	Water	93
Pressurizer/steam generator shield walls above operating floor**	Concrete	12
Fuel transfer canal wall	Concrete	54 to 72
Fuel transfer tube shielding	Concrete and steel	See Figures 12.3-17 and 12.3-18

TABLE 12.3-3 (Cont)

Shield Description	Material	Thickness (in)
Incore instrumentation cubicle wall	Concrete	39
Incore instrumentation source storage area	Concrete	42
Incore instrumentation access ***	Concrete	36
Steam generator cubicle wall	Concrete	36
Regenerative heat exchanger cubicle wall	Concrete	24
Cable vault wall	Concrete	24
Recirculation spray pump room in safeguard area	Concrete	36
Safeguards area wall	Concrete	24

NOTES:

* All concrete is reinforced with steel.

** The steam generator walls facing the equipment and personnel access hatches are 18 inches thick.

*** The drain opening in the incore instrumentation access wall is arranged to minimize direct line-of-sight radiation streaming through the opening, and is provided with a permanent 3 1/4" thick box-shaped steel shield at the outside of the opening.

TABLE 12.3-4

POST-ACCIDENT VITAL AREAS

No.	Description	Area or Route	Time after DBA	Occupancy Time of Vital Area	Calculated Whole Body Dose (Rem) Primary	Alternate
1	Control room (CR)	Area	0	Continuous to 30 days	2.16 Rem TEDE (Note 6)	NA
2	Emergency response facility	Area	0	Continuous to 30 days	3.44 Rem TEDE (Note 6)	NA
4	Operation of post-accident H_2 analyzer	Route	30 min	15 min	1.26	2.77
6	Lubrication oil for the diesel generator	Route	96 hrs	3 hr	0.01	Note 4
8	Personnel movement (emergency response facility to CR or CR to emergency response facility)	Route	Variable	3 min (walking) 1 min (riding)	Table 12.3-7	Note 4
9	Manual operations of valves in safeguards building	Route	24 hrs	15 min	3.15	3.40
10	Residual heat removal (RHR) suction valve transfer	Route	24 hrs	15 min	2.90	3.53
11.	Electrical connection of permanently installed spare equipment	Route	24 hrs	15 min	0.01	0.18
12	Ventilation fan for service water system	Route	24 hrs	30 min	0.01	Note 4
13	Obtaining and Analyzing a Post - Accident Effluent Sample	Route	1 hr	10 min (ERM) and 1 hr (lab)	4.4 [(2)] 4.3 [(3)]	NA NA
14	Re-energizing ECCS Valve Circuits	Route	25 min	5 min	0.40	0.59

TABLE 12.3-4 (Cont)

NOTES:

1. Security will be controlled at the secondary alarm station in the primary access facility, or as described in the site emergency plan.

2. Doses are calculated assuming the sample will be analyzed at the primary chemistry laboratory.

3. Doses are calculated assuming the sample will be analyzed at the emergency response facility.

4. Alternate route not considered because significant dose contribution is due to travel outside the buildings.

5. Not Used

6. Access areas 1 and 2, control room and Emergency Response Facility (ERF), have been evaluated for post-LOCA habitability based on use of the Alternative Source Term methodology of RG 1.183, and the associated 10 CFR 50.67 dose limit of 5.0 Rem TEDE.

TABLE 12.3-7

INTEGRATED DOSE FOR ONE WAY TRIP
BETWEEN CONTROL ROOM AND EMERGENCY RESPONSE FACILITY

Time After Start of DBA (hours)	Integrated Dose (Rem)
0	1.12
0.083	0.57
1	0.20
2	0.14
8	0.037
24	0.007
96	*
192	*
360	*
720	*

NOTE:

*Less than 1 mRem.

TABLE 12.3-8

VALVES POTENTIALLY REQUIRING REMOTE MECHANICAL OPERATION FOLLOWING A DBA

System	Valve Mark No.	Valve Function
Service Water System (SWS)	2SWS*MOV 104A 2SWS*MOV 104B 2SWS*MOV 104C 2SWS*MOV 104D	Service water inlet to recirculation spray heat exchangers
	2SWS*MOV 105A 2SWS*MOV 105B 2SWS*MOV 105C 2SWS*MOV 105D	Service water outlet from recirculation spray heat exchangers
	2SWS*104 2SWS*82	Inlet and outlet service water header cross connects to recirculation spray heat exchangers
Recirculation Spray System (RSS)	2RSS*MOV 156C 2RSS*MOV 156D	Recirculation spray pump discharge isolation
	2RSS*33 2RSS*35 2RSS*36 2RSS*34	Demineralized water inlet to recirculation spray pump seal system fill
Quench Spray System (QSS)	2QSS*MOV 101A 2QSS*MOV 101B	Quench spray pump discharge isolation
Safety Injection System (SIS)	2SIS*MOV 8889	Recirculation mode safety injection to discharge isolation
	2SIS*MOV 8811A 2SIS*MOV 8811B	Recirculation spray pump discharge to safety injection isolation

TABLE 12.3-9

AIRBORNE RADIATION MONITORS

Monitor	Quantity	Medium	Sensitivity (μCi/cc)	Minimum Range (decades)	Expected Concentrations	Location
Containment atmosphere (2 detectors)	1	Air			Table 12.2-24	Fig. 12.3-3
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		
Auxiliary building (2 detectors)	7	Air			Table 12.2-24	Figs. 12.3-3 and 12.3-4
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		
Waste handling building (2 detectors)	1	Air			Background	Fig. 12.3-5
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		
Fuel building vent (2 detectors)	1	Air			Background	Fig. 12.3-5
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		
Control room area (2 detectors)	1	Air			Background	Fig. 12.3-3
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		
Leak collection area (2 detectors)	1	Air			Background	Fig. 12.3-5
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		

TABLE 12.3-9 (Cont)

Monitor	Quantity	Medium	Sensitivity (μCi/cc)	Minimum Range (decades)	Expected concentrations	Location
Condensate polishing building	1	Air			Background	Fig. 12.3-6
(2 detectors)						
particulate			1×10^{-10}(I-131)	5		
gas			1×10^{-6}(Xe-133)	5		

TABLE 12.3-10

AREA RADIATION MONITOR LOCATIONS AND RANGES

Detector Location	Sensitivity and Range* (mRem/hr)
Reactor containment area, low range	$1\text{-}10^{5}$
Outside personnel hatch area	$0.1\text{-}10^{7}$
Reactor in-containment area, high range	$10^{3}\text{-}10^{10}$
Manipulator crane	$0.1\text{-}10^{4}$
In-core instrumentation area	$1\text{-}10^{5}$
Decontamination area	$0.1\text{-}10^{4}$
New fuel storage area	$0.1\text{-}10^{4}$
Fuel pit bridge	$0.1\text{-}10^{4}$
Auxiliary building	$0.1\text{-}10^{4}$
Sample room	$0.1\text{-}10^{4}$
Waste handling area	$0.1\text{-}10^{4}$
Condensate polishing area	$0.1\text{-}10^{4}$
Control room	$10^{-2}\text{-}10^{3}$
Safeguards recombiner area	$1\text{-}10^{5}$
Primary access facility	$0.1\text{-}10^{4}$

NOTE:

*Sensitivity is equal to the lower value of the range.

TABLE 12.3-11

EQUIPMENT SPECIFICATION LIMITS FOR COBALT IMPURITY LEVELS

Components	Material	Maximum Weight Percent Cobalt
Reactor internals (nonactive region)	SS*	0.20
Reactor internals (active region)	SS	0.12
Reactor vessel clad	SS	0.20
Reactor coolant piping	SS	0.20
Reactor internal bolting material	SS	0.25
Reactor coolant pumps	SS	0.20
Pressurizer	SS	0.20
Auxiliary heat exchanger surfaces exposed to reactor coolant	SS	0.20
Steam generators	Inconel	0.10
Fuel (nonactive region)	SS	0.12
Fuel (active region)	SS	0.08
Fuel	Inconel	0.10
Fuel	Zircoloy	0.002

*Stainless steel

TABLE 12.3-12

APPROXIMATE REACTOR COOLANT SYSTEM WETTED SURFACE AREAS

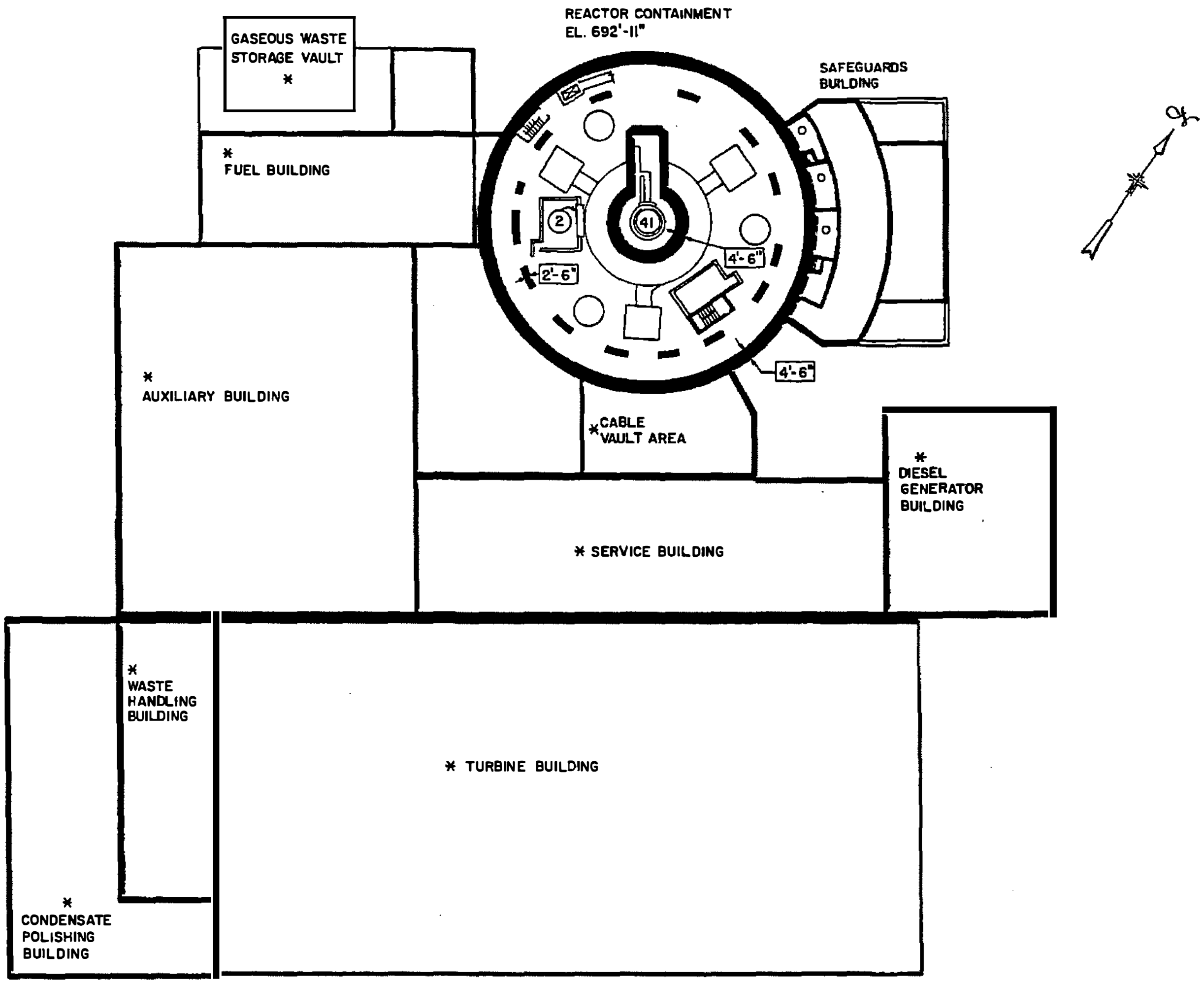
Component	Material	Surface Area (ft^2)
Reactor internals	SS*	4,326
Reactor vessel clad	SS	2,190
Reactor coolant piping	SS	2,750
Reactor internal bolting material	SS	Negligible
Reactor coolant pumps	SS	Negligible
Auxiliary heat exchanger surfaces	Inconel	Negligible
Steam generators	Inconel	1.90×10^5
Fuel (nonactive region)	SS	2,000
Fuel (active region)	SS	3,600
Fuel	Inconel	7.80×10^3
Fuel	Zircoloy	7.78×10^4

*Stainless steel

TABLE 12.3-13

APPROXIMATE REACTOR COOLANT SYSTEM WETTED SURFACE AREA OF STELLITE

Component	Surface Area (ft^2)
Reactor internals	3.2
Reactor coolant pipe journals	17.2
Control rod drive mechanisms	10.0
Reactor coolant system valves	2.6



LEGEND:

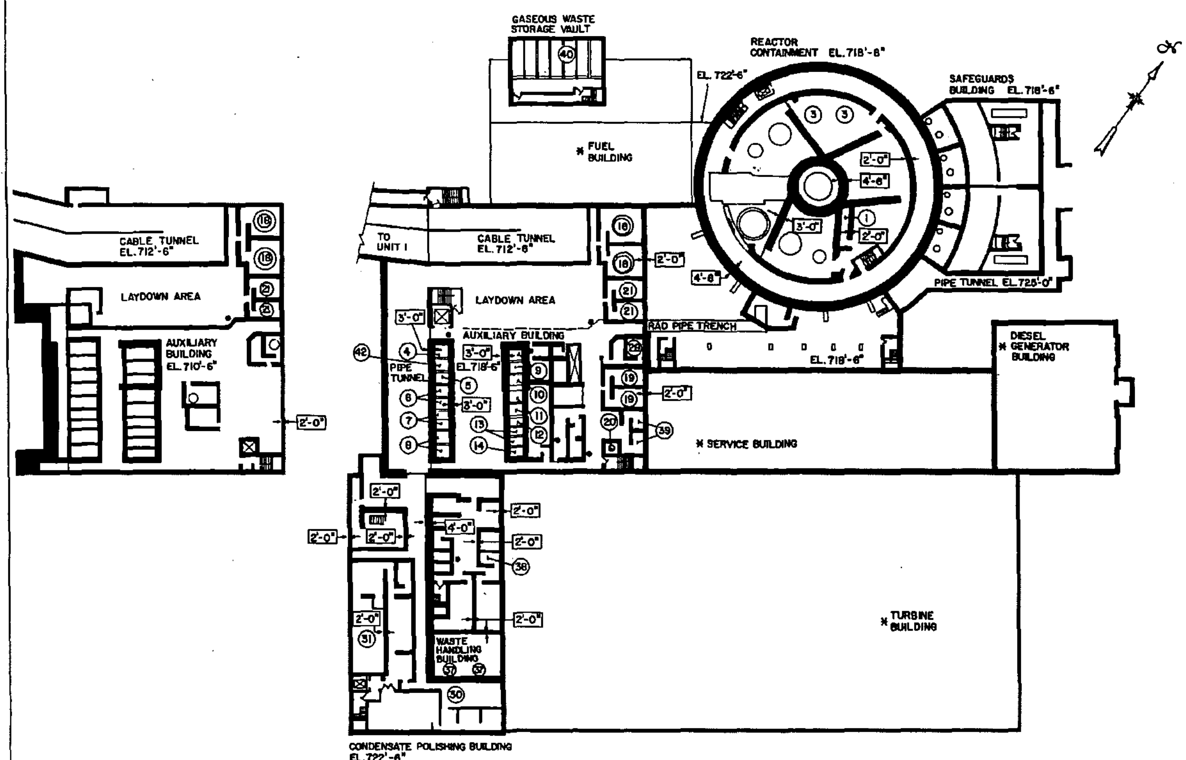
X'-X" — SHIELD WALL THICKNESS

MAJOR RADIATION SOURCES

EQUIPMENT NUMBER	CONTAINMENT BUILDING
2	EXCESS LETDOWN HEAT EXCHANGER
41	REACTOR CORE

NOTE:
* INDICATES BUILDING FOUNDATION OUTLINE ONLY

FIGURE 12.3-1
SHIELDING ARRANGEMENT AND FACILITIES
EL. 692'-11
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

X'-X" — SHIELD WALL THICKNESS (FEET-INCHES)

• AREA MONITOR

MAJOR RADIATION SOURCES

EQUIPMENT NUMBER	CONTAINMENT BUILDING
1	REGENERATIVE HEAT EXCHANGER
3	RESIDUAL HEAT REMOVAL HEAT EXCHANGER
	AUXILIARY BUILDING
4	BORIC ACID FILTER
5	FUEL POOL ION EXCHANGER
6	FUEL POOL FILTERS A & B
7	COOLANT RECOVERY FILTERS A & B
8	CESIUM DEMINERALIZERS A & B
9	MIXED BED DEMINERALIZERS A & B
10	CATION BED DEMINERALIZER
11	DEBORATING DEMINERALIZER A & B
12	SEAL WATER FILTER
13	SEAL WATER INJECTION FILTERS A & B
14	REACTOR COOLANT FILTER
15	LIQUID WASTE TANKS A & B
18	DEGASSIFIER RECIRCULATION PUMPS A & B
20	WASTE GAS SURGE TANK
21	LIQUID WASTE PUMPS A & B
28	PRIMARY DRAINS TRANSFER TANK
39	OVERHEAD GAS COMPRESSORS
42	CLEANUP FILTER
	CONDENSATE POLISHING BUILDING
30	CONDENSATE BACKWASH HOLD TANK
31	CONDENSATE BACKWASH FEED TANK
	WASTE HANDLING BUILDING
38	EVAPORATOR BOTTOMS HOLD TANK
	GASEOUS WASTE STORAGE VAULT
40	GASEOUS WASTE STORAGE TANKS

NOTE
* - INDICATES BUILDING OUTLINE ONLY

FIGURE 12.3-2
SHIELDING ARRANGEMENT AND FACILITIES
EL. 710'-6" TO 722'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-3
SHIELDING ARRANGEMENT AND FACILITIES
EL. 730'-6" TO 738'-10"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-4
SHIELDING ARRANGEMENT AND FACILITIES
EL. 745'-6" TO 755'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-5
SHIELDING ARRANGEMENT AND FACILITIES
EL. 760' TO 774'-6"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

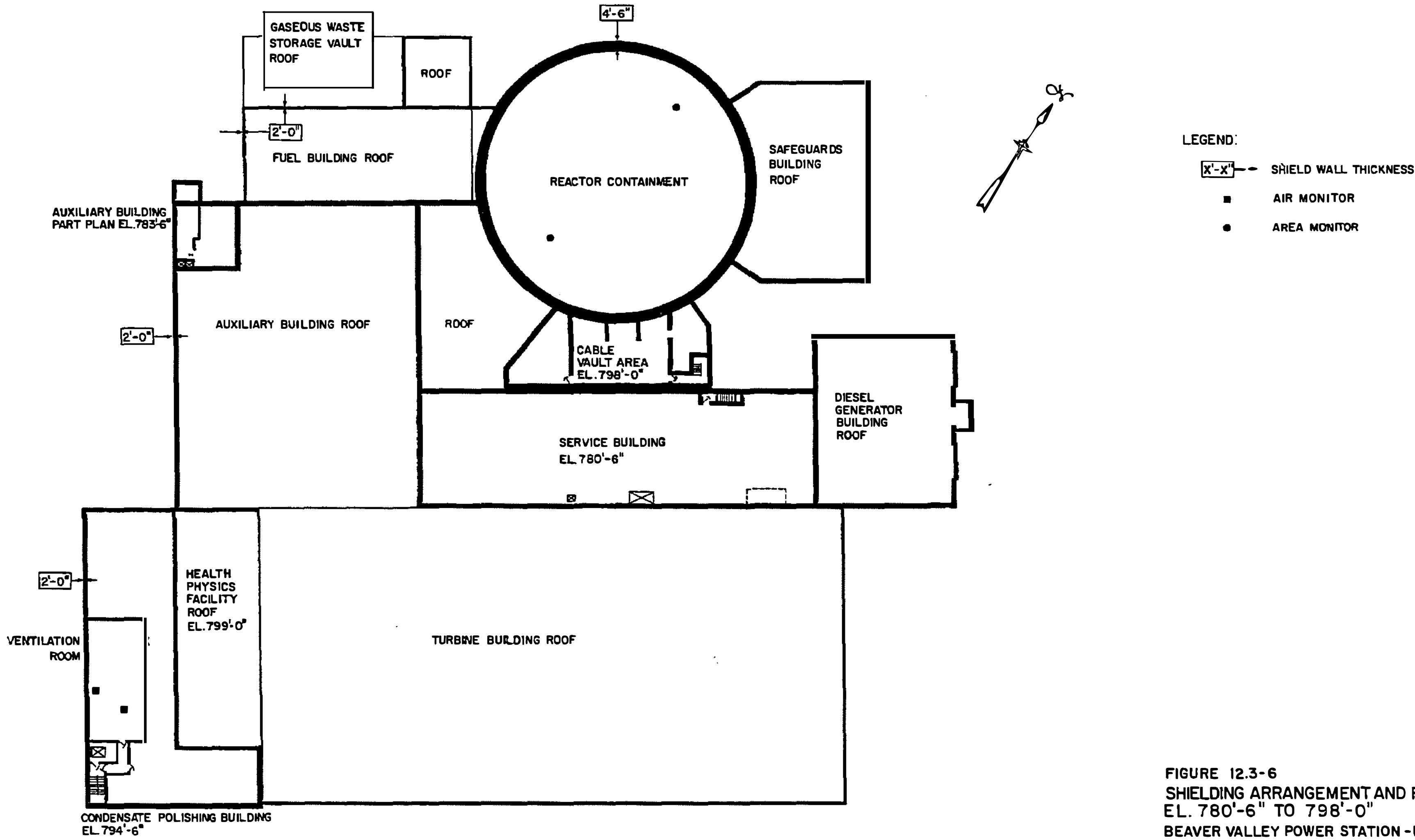
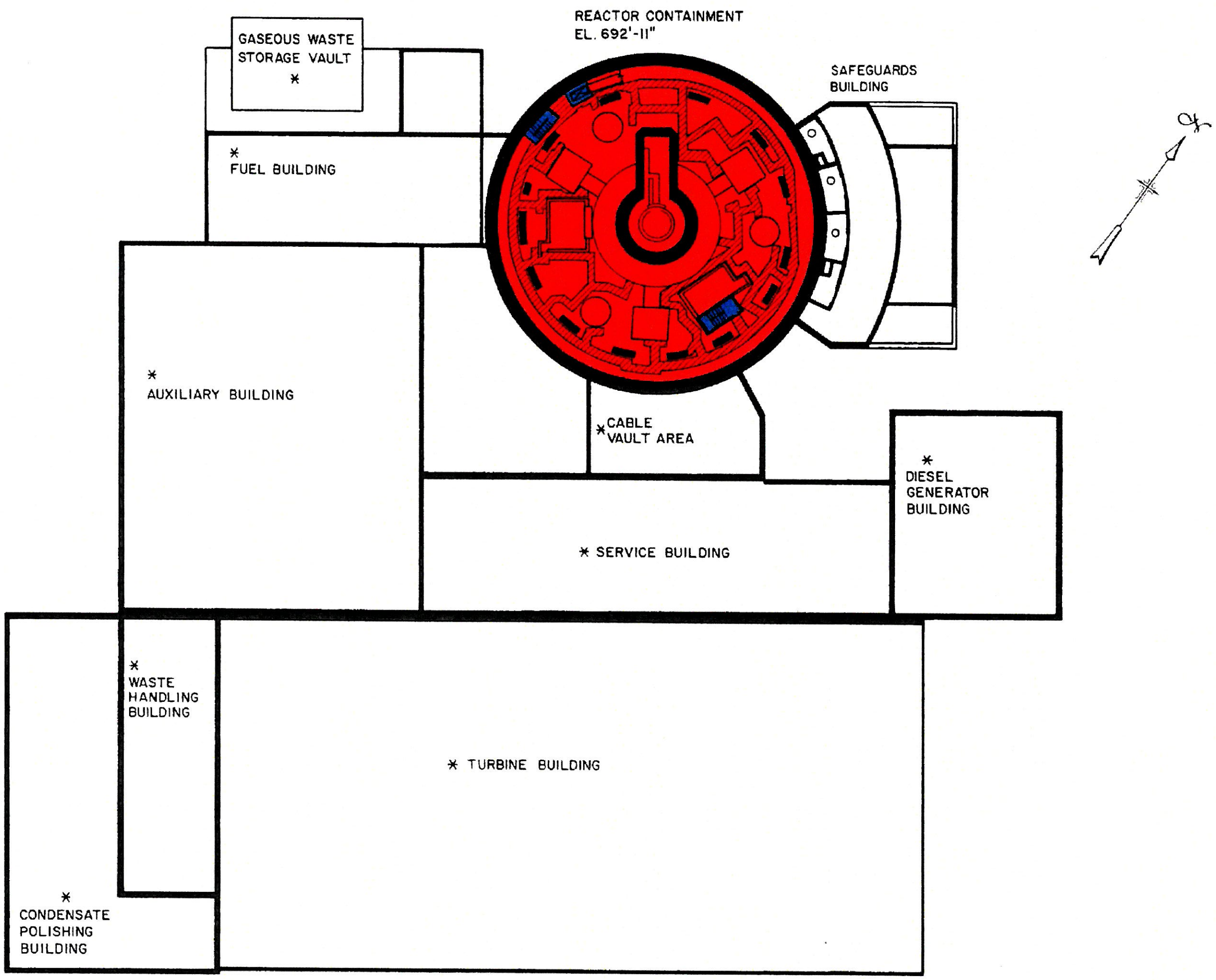


FIGURE 12.3-6
SHIELDING ARRANGEMENT AND FACILITIES
EL. 780'-6" TO 798'-0"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

ZONE	COLOR CODE	EXPOSURE RATE
V		≥ 100 MREM/HR
IV		< 100 MREM/HR
III		≤ 15 MREM/HR
II		≤ 2.5 MREM/HR
I		≤ 0.75 MREM/HR

WALKWAY AND EQUIPMENT REMOVAL AISLE

NOTE:
* INDICATES BUILDING FOUNDATION OUTLINE ONLY

FIGURE 12.3-7
RADIATION ZONES
EL. 692'-11
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

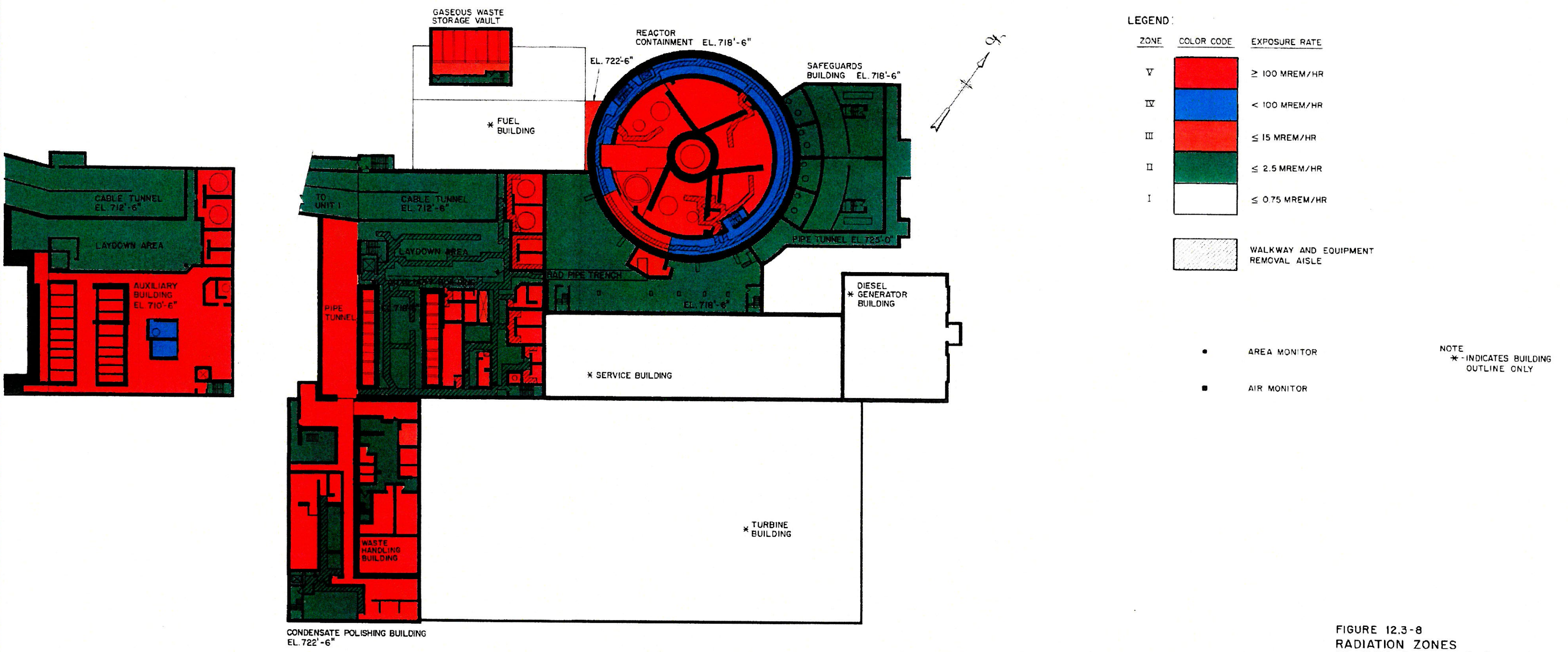


FIGURE 12.3-8
RADIATION ZONES
EL. 710'-6" TO 722'-6"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-9
RADIATION ZONES
EL. 730'-6" TO 738'-10"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-10
RADIATION ZONES
EL. 745'-6" TO 755'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 12.3-11
RADIATION ZONES
EL. 760' TO 774'-6"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

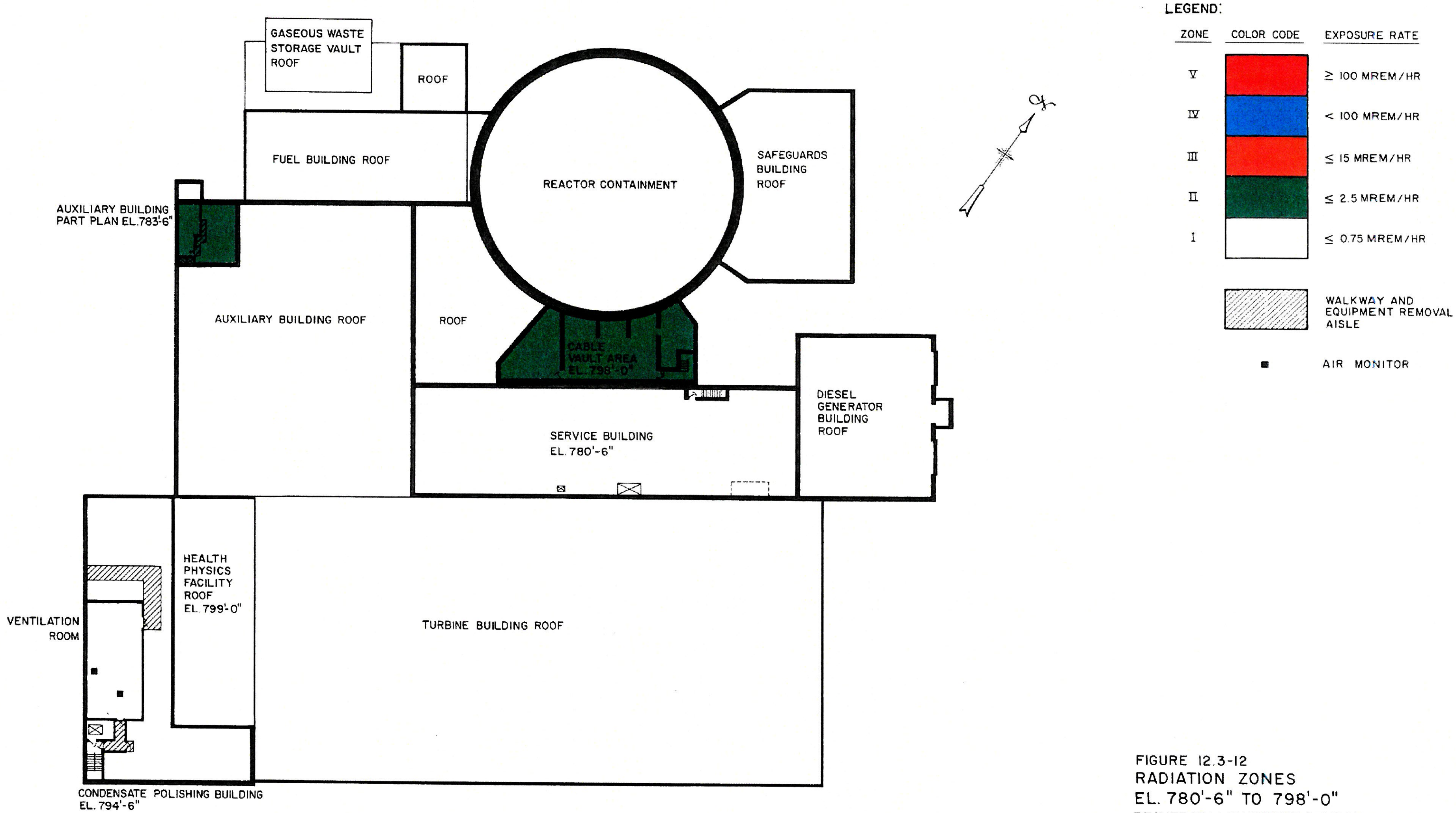
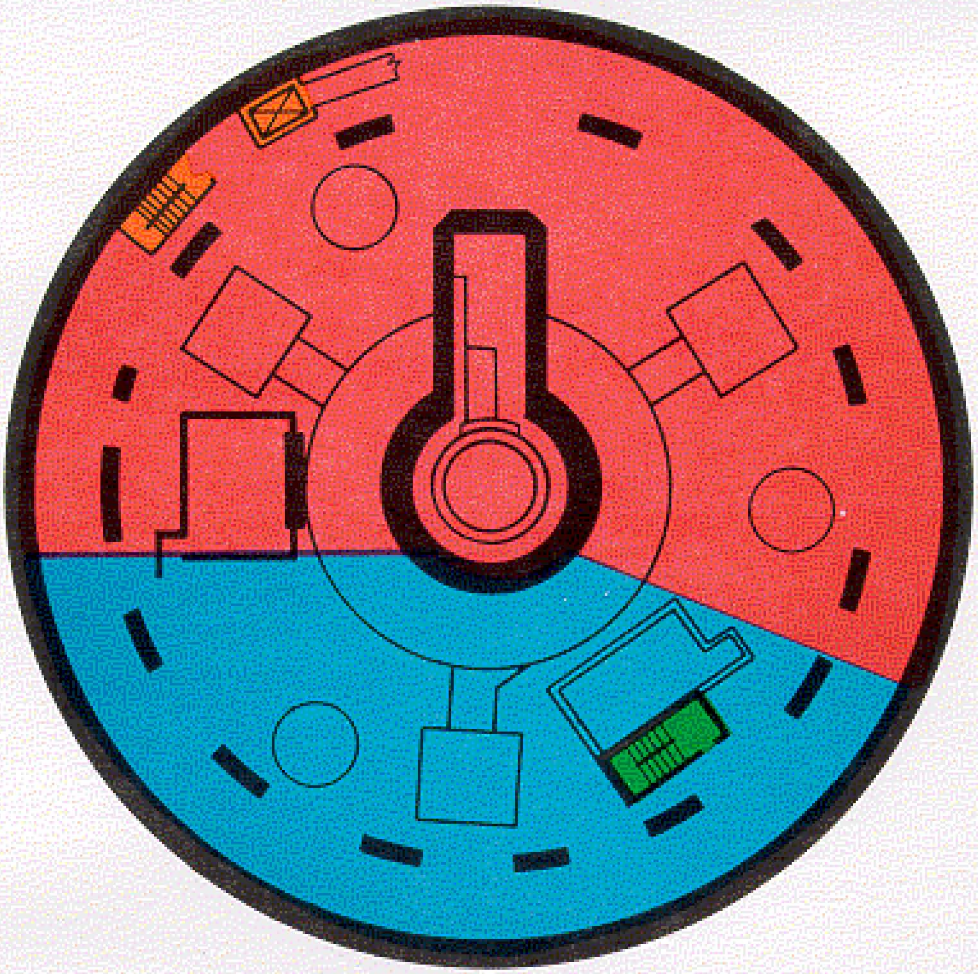


FIGURE 12.3-12
RADIATION ZONES
EL. 780'-6" TO 798'-0"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

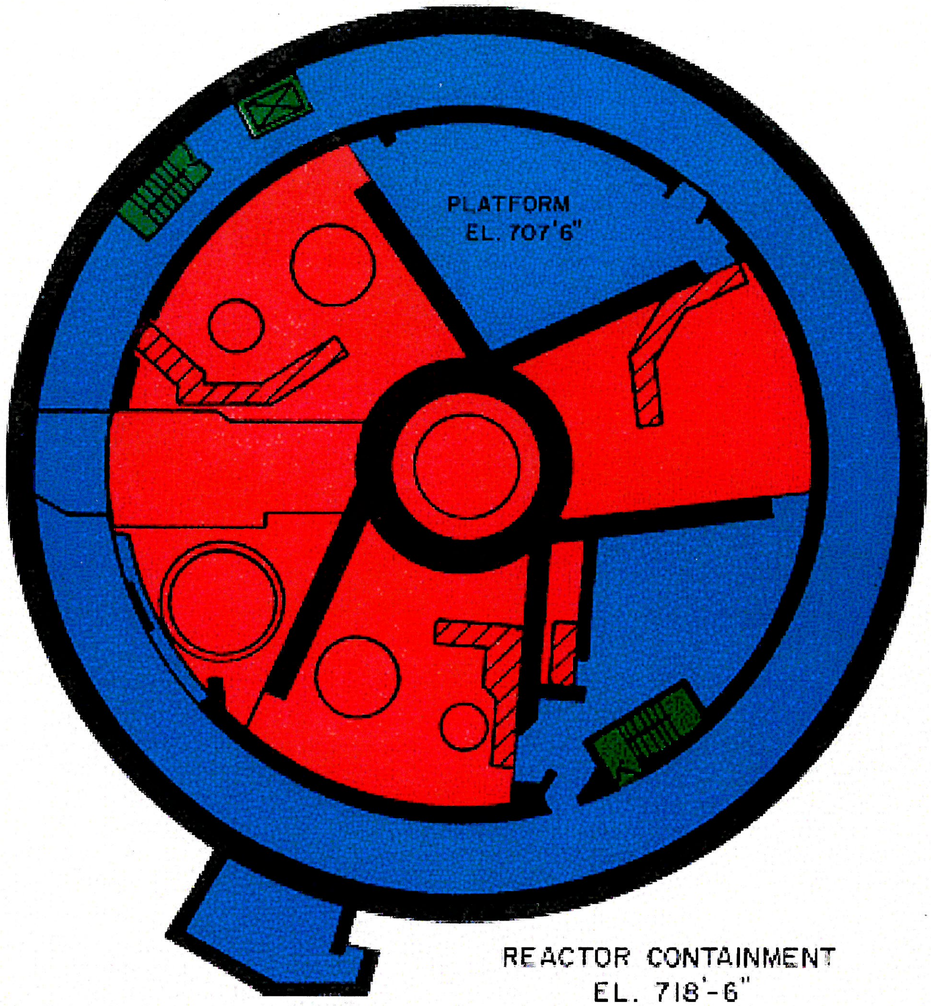


REACTOR CONTAINMENT - EL. 692'-11"

LEGEND:

ZONE	COLOR CODE	EXPOSURE RATE
V	(red)	≥ 100 MREM/HR.
IV	(blue)	< 100 MREM/HR.
III	(orange)	≤ 15 MREM/HR.
II	(green)	≤ 2.5 MREM/HR.
I	(white)	≤ 0.75 MREM/HR.

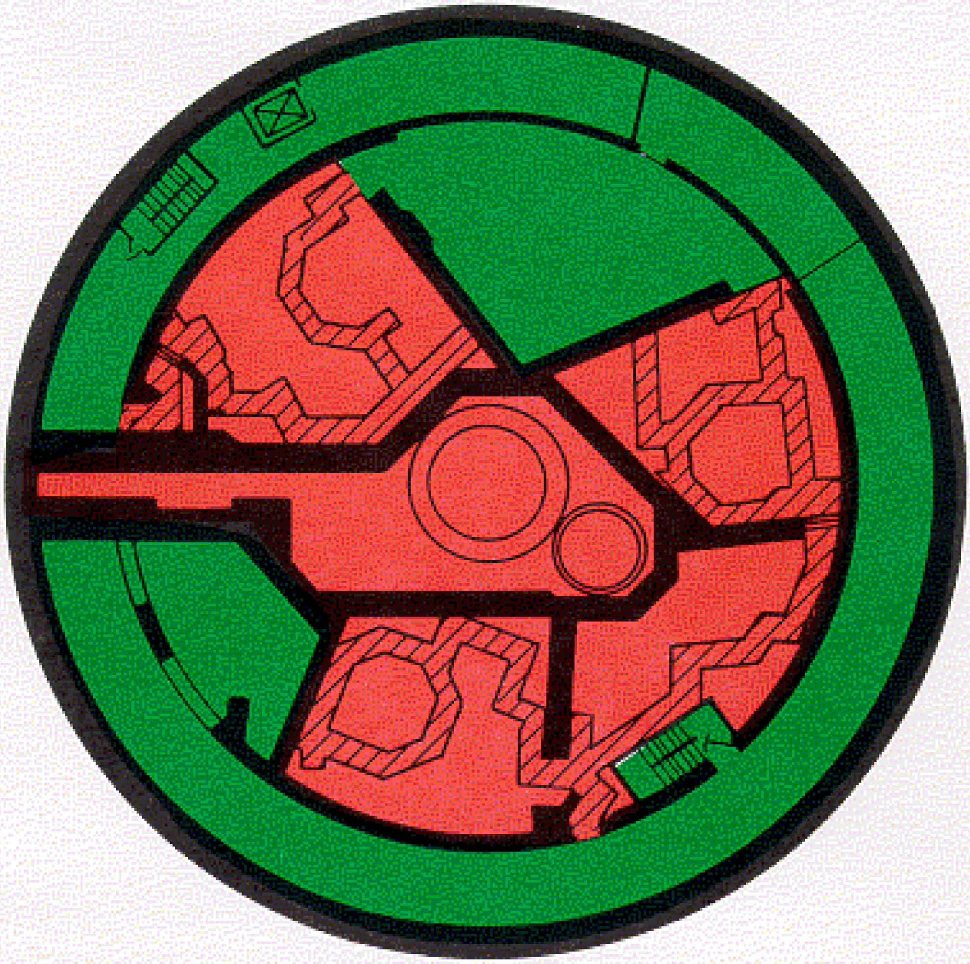
FIGURE 12.3 - 13
RADIATION ZONES-SHUTDOWN/REFUELING
EL. 692'-11"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

ZONE	COLOR CODE	EXPOSURE RATE
V	(red)	≥ 100 MREM/HR.
IV	(blue)	< 100 MREM/HR.
III	(orange)	≤ 15 MREM/HR.
II	(green)	≤ 2.5 MREM/HR.
I	(white)	≤ 0.75 MREM/HR.
	(hatched)	WALKWAY AND EQUIPMENT REMOVAL AISLE

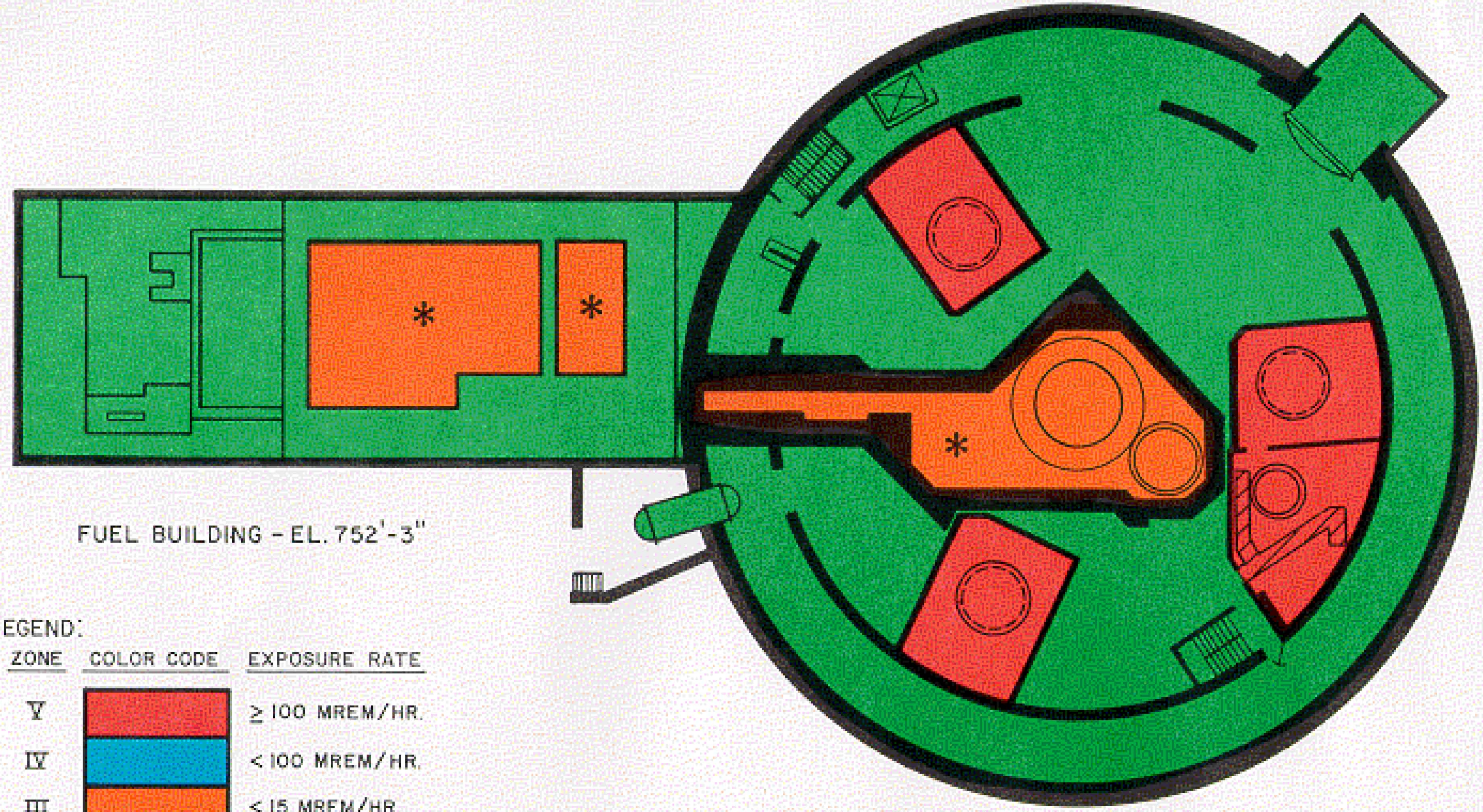
FIGURE 12.3-14
RADIATION ZONES-SHUTDOWN/REFUELING
EL. 718'-6"
BEAVER VALLEY POWER STATION-UNIT 3
FINAL SAFETY ANALYSIS REPORT



REACTOR CONTAINMENT - EL. 738'-0"

ZONE	COLOR CODE	EXPOSURE RATE
V		≥ 100 MREM/HR.
IV		< 100 MREM/HR.
III		≤ 15 MREM/HR.
II		≤ 2.5 MREM/HR.
I		≤ 0.75 MREM/HR.
		WALKWAY AND EQUIPMENT REMOVAL AISLE

FIGURE 12.3-15
RADIATION ZONES-SHUTDOWN/REFUELING
EL. 738'-0"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

ZONE	COLOR CODE	EXPOSURE RATE
V		≥ 100 MREM/HR.
IV		< 100 MREM/HR.
III		≤ 15 MREM/HR.
II		≤ 2.5 MREM/HR.
I		≤ 0.75 MREM/HR.
		WALKWAY AND EQUIPMENT REMOVAL AISLE

* DOSE RATE TO OPERATING PERSONNEL IS ≤ 15 MREM/HR. FROM RAISED SPENT FUEL ASSEMBLY.

FIGURE 12.3-16
RADIATION ZONES-SHUTDOWN/REFUELING
EL. 752'-3" TO 767'-10"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

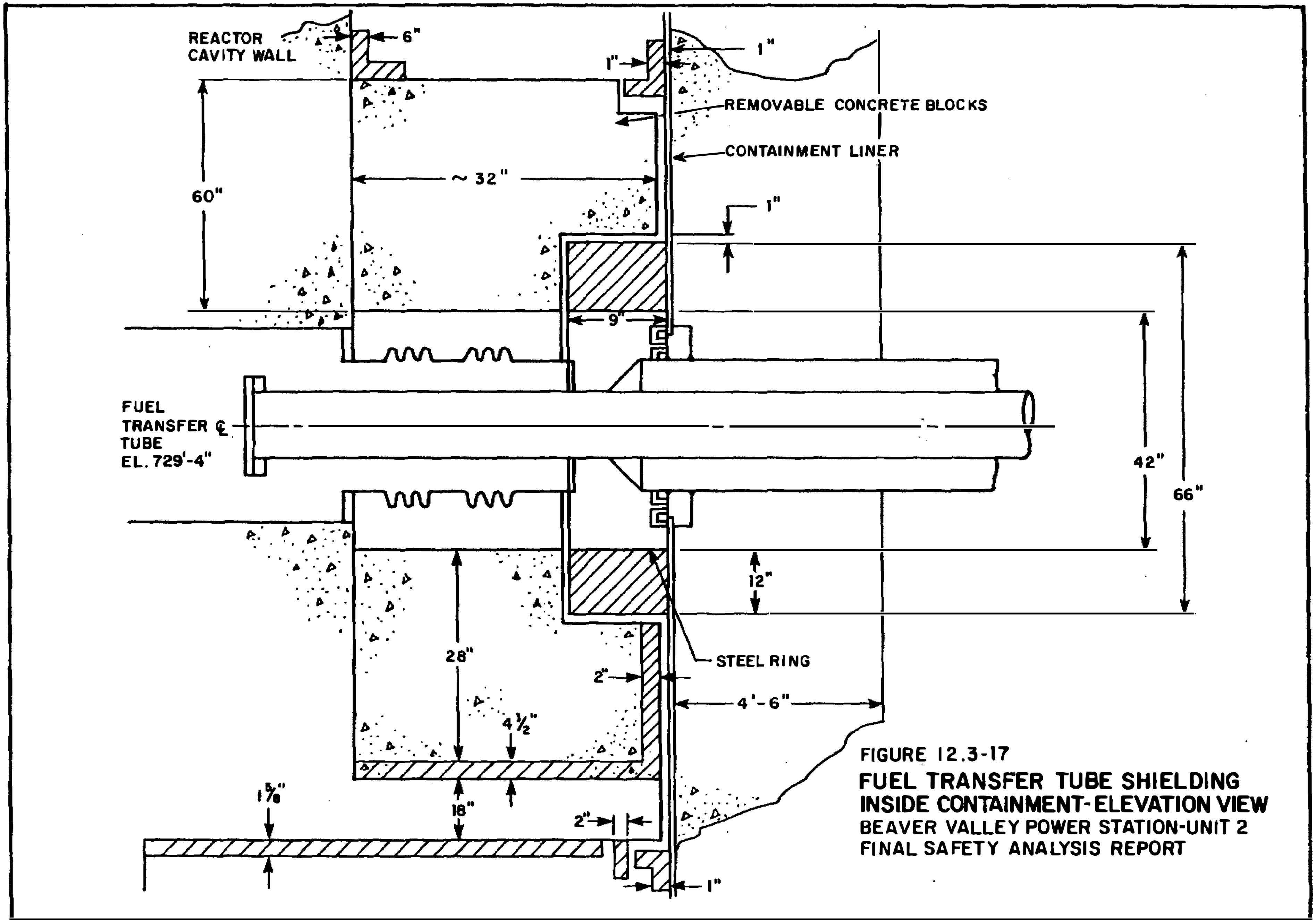


FIGURE 12.3-17
FUEL TRANSFER TUBE SHIELDING INSIDE CONTAINMENT-ELEVATION VIEW
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

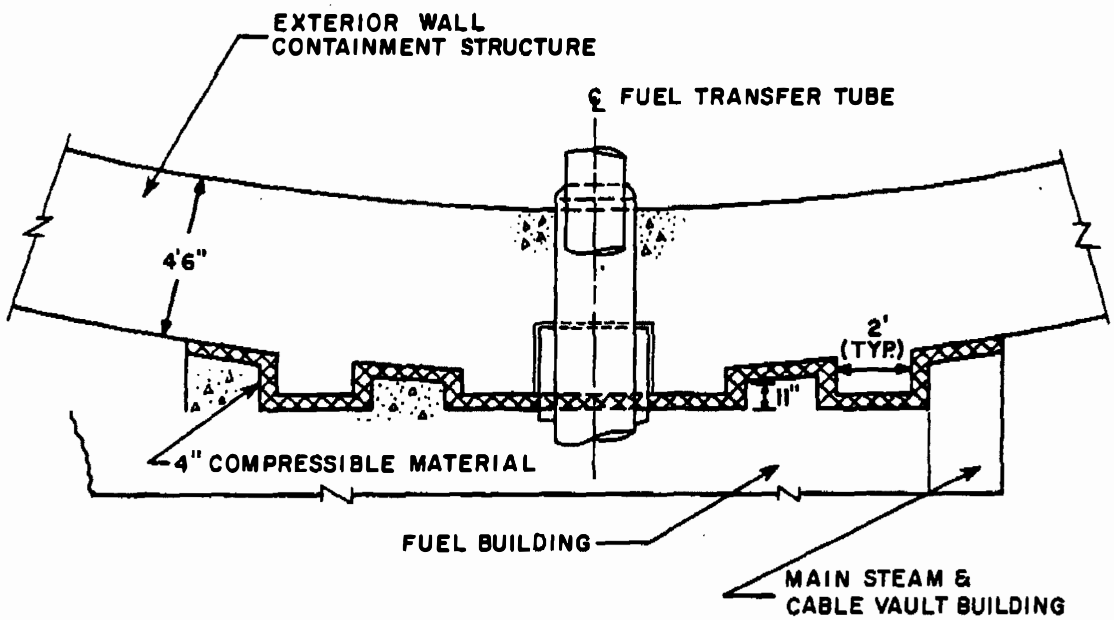


FIGURE 12.3-18
FUEL TRANSFER TUBE SHIELDING OUTSIDE CONTAINMENT- PLAN VIEW
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

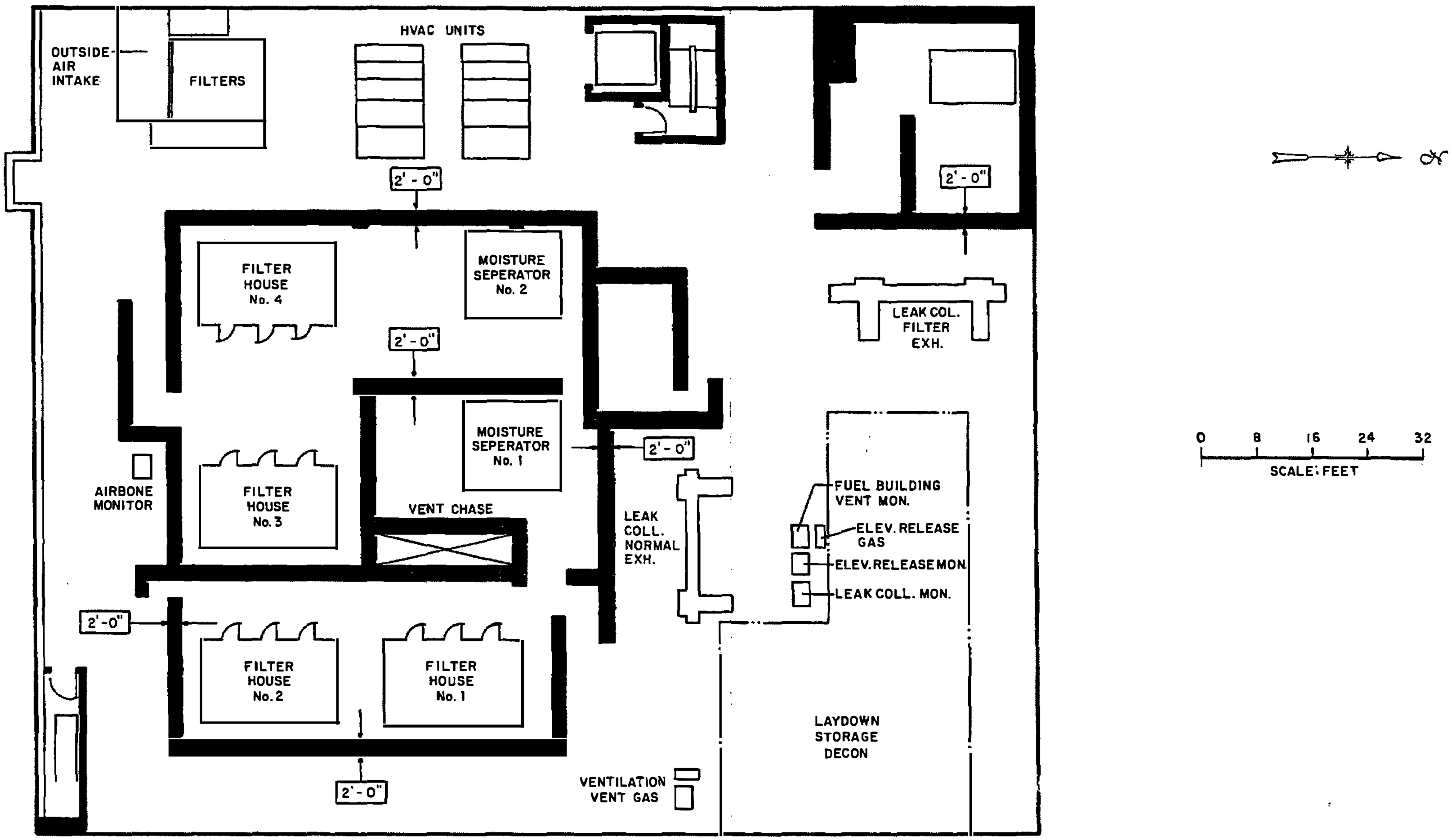


FIGURE 12.3-32
AIR CLEANING SYSTEM HOUSING
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

12.4 DOSE ASSESSMENT

12.4.1 Estimates of Exposure

The occupational radiation dose and number of personnel required at a pressurized water reactor (PWR) are mostly functions of the degree of routine and special maintenance performed in a given year. Table 12.4-1 presents a summary for years 1969-1980 as reported by the U.S. Nuclear Regulatory Commission (USNRC 1981). Tables 12.4-2 and 12.4-3 show 1980 collective annual doses by work function and by occupation, respectively, for PWRs (USNRC 1981). The average exposure was approximately 450 man-Rem per reactor-year including all workers (such as utility personnel, vendor personnel, and contractor personnel). Approximately 70 percent of this exposure is estimated to be associated with routine and special maintenance activities. To minimize this exposure, plant shielding and machinery locations have been designed to provide maximum laydown space, maximum working room, and minimum time required to perform operations. Experience gained in the operation of nuclear plants is factored into these designs with the objective of minimizing total exposure to Beaver Valley Power Station - Unit 2 (BVPS-2) personnel in agreement with the objectives of Regulatory Guide 8.8. Section 12.3 describes radiation protection design features.

Table 12.4-4 presents estimates for BVPS-2 of total man-Rem percentage and man-Rem dose for various work functions. The values listed are estimates based upon the percentages and average man-Rem per reactor-year given in NUREG-0713 (USNRC 1981).

Table 12.4-5 shows an exposure breakdown by system at one PWR plant, and Table 12.4-6 presents an exposure breakdown by work group for one year at another PWR plant (EPRI 1981).

Operating experience for PWRs indicates that subtracting out special maintenance activities results in relatively constant exposures, after an initial buildup during the first 2 years attributable primarily to Co-58 which has a half life of 71 days. After Co-58 has reached an equilibrium condition, both Co-58 and Co-60 are believed to contribute substantially to exposure levels. The relatively constant exposure resulting from routine work at three PWR plants is presented in Table 12.4-7. Special maintenance activities, such as steam generator repairs, have caused major fluctuations in man-Rem doses at PWRs.

The above information was developed in support of the original license and is retained here for historical purposes.

In addition to special maintenance, BVPS-2 operation, supervision, and radiation protection activities will affect the total man-Rem. The timing of work and special tests will be arranged for times of lowest dose rates consistent with reasonable operation of BVPS-2. Trained personnel in the Radiological Control staff will control access to high radiation areas and provide continuous surveillance of personnel exposure.

12.4.1.1 Estimates of Dose to Construction Workers

The estimated dose to construction workers at BVPS-2 due to the operation of Beaver Valley Power Station - Unit 1 (BVPS-1) is shown in Table 12.4-8. The table presents the expected number of construction workers from 1981 through the middle of 1986. Also presented is the estimated yearly dose to the construction workers at the BVPS-2 site due to radiation from BVPS-1 operation. The dose was estimated based upon readings from selected thermoluminescent dosimeters (TLD) on the industrial fence separating the BVPS-2 construction site from the BVPS-1 site and other selected locations around the BVPS-2 site. The selected TLD readings were considered representative of the dose rate which the construction workers would receive. Based upon available data for the year 1981, Table 12.4-8 shows the estimated dose to construction workers through the middle of 1986. These estimated doses are based on an expected capacity factor of 0.8 for BVPS-1 from 1981 through 1986. This information was developed in support of the original license and is retained here for historical purposes.

12.4.2 Estimates of Inhalation Thyroid Doses

Inhalation doses will be negligible in every area except the containment, turbine, and auxiliary building areas. Potential airborne activities for these areas are given in Section 12.2. These concentrations are based on expected failed fuel and leakage assumptions as given in NUREG-0017 (USNRC 1976). The inhalation thyroid doses that result are given in Table 12.4-9. This information was developed in support of the original license and is retained here for historical purposes.

Thyroid dose rates in Table 12.4-9 are calculated according to:

$$D_t = \sum_i (B.R.)(A_i)(C)(K_i)$$

where:

D_t = thyroid dose rate (mRem/hr)

K_i = thyroid dose conversion factor (from TID 14844, March 1962) (Rem/Ci)

B.R. = breathing rate, m^3/sec

A_i = building airborne concentration, Table 12.2-24, of the ith isotope (μCi/cm^3)

C = 3,600 sec/hr x 10^3 mRem/Rem.

12.4.3 Estimates of Site Area Boundary Doses

Site boundary doses are based on radioactive gaseous effluents from BVPS-1 and BVPS-2, as given in Tables 11.3-2 and 11.3-3, respectively. Table 12.4-10 gives the doses at the worst case location. This information was developed in support of the original license and is retained here for historical purposes. Note that for an operating plant such as BVPS, actual dose information is provided in the Annual Radioactive Effluent Reports.

12.4.4 References for Section 12.4

Electric Power Research Institute (EPRI) 1981. Occupational Radiation Exposure Radiation Technology Planning Study. EPRI NP-1862 TPS 79-761.

U.S. Atomic Energy Commission (USAEC) 1962. Calculation of Distance Factors for Power and Test Reactor Sites. TID-14844, Division, of Technical Information.

U.S. Nuclear Regulatory Commission (USNRC) 1976. Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors. Office of Standards Development, NUREG-0017.

USNRC 1981. Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1980. Office of Management and Program Analysis, NUREG-0713, Vol 2.

Tables for Section 12.4

TABLE 12.4-1 (Historical)

MAN-REM SUMMARY FOR PRESSURIZED WATER REACTORS*

Year	Number of Operating Reactors	Average Rated Capacity (MWe)	Average MW-Year Generated	Yearly Average Man-Rem/ Reactor-Year	Yearly Average Man-Rem/ MW-Year
1969	4	349	274	165	0.6
1970	4	349	245	684	2.8
1971	6	399	319	307	1.0
1972	8	446	318	464	1.5
1973	12	533	314	783	2.5
1974	20	619	341	331	1.0
1975	26	643	461	318	0.7
1976	30	675	444	460	1.0
1977	34	699	510	396	0.8
1978	39	723	509	429	0.8
1979	42	729	434	510	1.2
1980	42	721	435	578	1.3

NOTE:

*USNRC 1981.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-2 (Historical)

1980 COLLECTIVE ANNUAL DOSES BY WORK FUNCTION*

	Pressurized Water Reactors	
Work Function	Man-Rem	% of Total
Reactor operations and surveillance	2,702	11.5
Routine maintenance	6,350	27.0
Inservice inspection	1,923	8.2
Special maintenance	10,277	43.6
Waste processing	630	2.6
Refueling	1,654	7.1
Totals	23,536	100.0

NOTE:

*USNRC 1981.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-3

1980 COLLECTIVE ANNUAL DOSES BY OCCUPATION* (Historical)

	Pressurized Water Reactors	
Occupation Category	Man-Rem	% of Total
Maintenance	12,763	69.6
Operations	1,217	6.6
Radiation Protection	2,007	10.9
Supervisory	439	2.4
Engineering	1,919	10.5
Totals	18,345	100.0

NOTE:

*USNRC 1981.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-4

ESTIMATES OF PERCENTAGE OF TOTAL MAN-REM AND COLLECTIVE DOSE EQUIVALENTS FOR VARIOUS WORK FUNCTIONS (Historical)

Work Function	Percentage of Total Man-Rem	Man-Rem/ Reactor/Year
Reactor operations and surveillance	11.5	52
Routine maintenance	27.0	122
Inservice inspection	8.2	37
Special maintenance	43.6	196
Waste processing	2.6	12
Refueling	7.1	32
Total	100.0	450

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-5

EXPOSURE BREAKDOWN BY SYSTEM FOR ONE YEAR
OF OPERATION AT POWER PLANT A* (Historical)

System	Man-Hours	Man-Rem	mR/hr from RWPs	mRem/ Man-Hour
Steam generator systems	2,405	64.36	26.6	26.8
Blanket radiation work permits (Primarily hp & operations)	15,892	61.24	3.5	3.9
Undefined systems	8,112	47.00	5.8	5.8
Reactor coolant system	2,965	39.74	13.4	13.4
Reactor vessel & appurtenances	1,934	29.68	15.2	15.3
Reactor control systems	3,491	16.29	4.7	4.7
CVCS/liquid poison system	1,668	13.13	7.8	7.9
Safety-related display instruments	1,168	12.33	10.4	10.6
Emergency core cooling system	1,120	9.44	8.3	8.3
Clean liquid waste process system	1,084	5.81	4.3	5.4
Feedwater systems	4,424	4.94	1.1	1.1
Fuel handling systems	1,061	4.46	4.2	4.2
Solid radwaste management systems	533	4.13	6.3	7.8
Reactivity control systems	444	3.74	8.4	8.4
Other coolant subsystems & controls	316	2.09	6.3	6.6
Liquid radwaste management systems	133	2.00	14.9	15.1
Aerated waste processing system	185	1.72	9.2	9.3
Reactor vessel internals	106	1.63	12.5	15.4
Spent fuel storage facilities	1,029	1.50	1.5	1.5
Reactor trip systems	397	1.50	3.2	3.8
Main steam systems	690	1.42	2.1	2.1
HVAC Systems	.90	1.00	5.3	5.3
Floor drains waste systems	92	0.97	9.1	10.5
Containment heat removal systems	179	0.81	4.5	4.5
Other engineered safeguards & controls	262	0.79	2.4	3.0
Fire protection	253	0.47	1.5	1.8
Other instrument systems for safety	23	0.48	10.4	21.3
Cooling systems reactor auxiliary	159	0.47	2.9	3.0
Containment air purification/-cleanup	98	0.41	4.0	4.2
Other systems	77	0.37	4.5	4.8
Compressed air systems	27	0.43	10.0	16.2
Steam generator blowdown	64	0.33	4.9	4.9
Process sampling systems	46	0.34	7.4	7.4
Engineered safeguards instrument systems	93	0.31	2.9	3.3
Spent fuel pool cooling & cleanup	53	0.21	4.0	4.0
Gaseous radwaste management system	91	0.22	2.4	2.4

TABLE 12.4-5 (Cont)

System	Man-Hours	Man-Rem	mR/hr from RWPs	mRem/ Man-Hours
Other instruments nonsafety	37	0.21	5.4	5.7
Residual heat removal system	18	0.17	3.5	9.5
Turbine generators	40	0.12	3.2	3.0
Containment isolation systems	29	0.12	5.9	4.1
AC onsite power systems	9	0.05	2.2	5.5
New fuel storage facilities	117	0.04	0.2	0.4
Demineralized water makeup systems	11	0.04	2.2	4.0
Communications systems	25	0.05	0.8	2.0
Onsite power systems ac/dc	10	0.04	-	3.8
Process & effluent radiation monitors	7	0.03	2.8	4.2
Other auxiliary process systems & controls	8	0.03	2.6	3.3
Containment combustible gas control	14	0.02	1.1	1.8
Airborne radiation monitor system	6	0.02	-	2.5
Other electric power systems & controls	11	0.02	1.4	1.4
Area radiation monitor system	53	0.01	-	0.1
Condensate & feedwater systems	4	0.01	2.9	2.9
Failed fuel detection systems	2	0.01	6.0	6.0
Condensate storage facilities	4	0.00	-	1.3
Other auxiliary water systems & controls	8	0.01	1.3	1.3
Equipment drains waste systems	3	0.01	-	1.7
Emergency lighting systems	2	0.00	-	-
Extraction & auxiliary steam systems	26	0.00	-	-
Total	51,307	336.77	-	6.5 (average)

NOTE:

*EPRI 1981.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-6

EXPOSURE BREAKDOWN BY WORK GROUP FOR ONE YEAR AT PWR PLANT B*
(Historical)

Work Group Classification	Total Dose (man-Rem)**	Number of Individuals	Average Dose Per Person (Rem)
Maintenance	408	301	1.36
Operations	58	54	1.07
Boiler maker	19	17	1.12
Custodial	25	27	0.91
Radiation protection	35	48	0.74
Inspector	38	64	0.59
Radiation chemistry	7	16	0.44
General labor	4	11	0.36
Engineering support	78	230	0.34
Instrumentation & calibration	11	31	0.34
All other	18	52	0.34
Clerical or stockroom	12	38	0.31
General vendor	1	20	0.28
Pipefitter	7	34	0.21
Radwaste processing & shipment	1	2	0.17
Carpenter	1	8	0.14
Construction	5	40	0.12
Electrical	8	48	0.12
Consultant	1	17	0.05
Security	1	124	0.01
Total	738***	1,182	0.62

NOTES:

*EPRI 1981.
**Rounded to the nearest integer.
***Represents 95% of total plant exposures.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-7

COMBINED EXPOSURE FROM ROUTINE WORK-1978
FOR THREE POWER PLANTS* IN 1978 (Historical)

Job Function	Exposure (man-Rem)
Station surveillance inspection and operation	170
Station maintenance	108
Radioactive waste handling and disposal	49
General cleanup and decontamination	28
Spent fuel handling and shipping	21
Filter change operation and disposal	19
Miscellaneous nuclear station modifications	7
Post irradiation examination	3
Total	405

Work Group	Exposure (man-Rem)
Maintenance	115
Operations	91
Radiation Protection	50
Engineering	35
Janitorial	29
Instrumentation and electrical	26
Chemistry and radiochemistry	24
Miscellaneous	17
Performance engineering	10
Supervision	5
Quality assurance	3
Total	405

NOTE: *EPRI 1981
The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 12.4-8 [HISTORICAL]

ESTIMATED ANNUAL DOSE TO BVPS-2 CONSTRUCTION WORKERS
DUE TO BVPS-1 OPERATION

Year	Yearly Number of Workers	Dose (man-Rem)
1981	825	5.7
1982	1,350	9.3
1983	1,500	10.3
1984	900	6.2
1985	250	1.7
1986 (6 mo)	100	0.3

The information presented above was developed in support of the original license and is retained here for historical purposes.

TABLE 12.4-9 [HISTORICAL]

ESTIMATES OF INHALATION THYROID DOSE RATES
IN MAJOR BUILDINGS

Building	Expected Dose Rate (mRem/hr)
Containment*	$2.5x10^{-1}$
Auxiliary	$3.7x10^{-1}$
Turbine	$1.9x10^{-4}$

NOTE:

*After 16 hours operation of containment recirculation filters with 2 fans. If only 1 fan is operating, the expected dose rate would be 7.2 mrem/hr.

The information presented above was developed in support of the original license and is retained here for historical purposes.

TABLE 12.4-10 [HISTORICAL]

SITE BOUNDARY DOSES

Radiation Source	Site Boundary* Dose Rate (mRem/yr)
Gaseous gamma	6.3
Gaseous beta	6.8
Direct	2.7

NOTE:

*Average annual dose rate at the security area boundary from TLD measurements = 30 mRem/yr.

The information presented above was developed in support of the original license and is retained here for historical purposes.

12.5 RADIATION PROTECTION PROGRAM

The radiation protection program at the Beaver Valley Power Station (BVPS) is developed and implemented to evaluate and document plant radiological conditions and assure that every reasonable effort is expended to maintain personnel exposure as low as reasonably achievable (ALARA). The radiation protection procedures will provide all requirements, criteria, and guidelines for the radiation protection program. This will include a description of the program objectives and the general responsibilities of all groups and individuals. Specific requirements such as those approved for radiation and contamination survey methods (including documentation and area posting) will be described in detail in these procedures.

The radiation protection program will be developed in accordance with 10 CFR 20 and other applicable federal and state regulations. Periodic reviews will ensure the manual is maintained current. Revisions in any applicable regulations that are adopted and published in the Federal Register or are issued in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides will be evaluated. If warranted, radiation protection procedures will be revised to be consistent with the revised rules and practices.

The radiation protection program and its implementation will be audited periodically by qualified person(s), who are not assigned to the station, to assure it is in compliance with existing requirements. Audit findings will be reported to management for their information and appropriate action.

The basic objective of the radiation protection program is to establish and enforce observance of radiological limits and policies which fully comply with the requirements of all current federal and state regulations. The program is also intended to effect standards and institute practices which will result in conservatism; thus minimizing significant radiological occurrences. In addition, the program will be carefully and conservatively implemented in order that radiation exposures and contamination levels will be as low as reasonably achievable consistent with sound operation and maintenance principles.

12.5.1 Organization and Responsibilities

The radiation protection organization acts in an advisory and support capacity to other site organizations. The organization is developed and the responsibilities, qualifications, and training of the radiation protection personnel are established to reflect the intent of Regulatory Guides 1.8, 8.2, 8.8, and 8.10. Radiation protection staff will meet the minimum experience and qualification requirements of ANSI N18.1-1971.

The responsibility for maintenance of safe radiological conditions and implementation of the radiation protection program is assigned to all individuals involved in the maintenance and operation of BVPS. Specific responsibilities relevant to the radiation protection organization are described below:

The radiation protection organization is positioned within the overall BVPS organization structure as described in Section 13.1. The radiation protection organization is composed of a manager who has several supervisors as direct reports. The organization's responsibilities are divided among the supervisors who maintain staffs of technicians and other support personnel to accomplish assigned tasks. Responsibilities include: implementing the ALARA and radwaste management programs; directing the implementation of all radiological control activities at BVPS; participating in the preparation, revision, and implementation of the emergency preparedness plan; advising and assisting plant management on matters concerning radiological safety of station personnel.

Typical functions in support of these responsibilities include: maintenance of relevant parts of radiation protection procedures; external and internal dose monitoring; implementing the respiratory protection program; preparation of occupational dose monitoring records and reports; engineering calculations; radiological shielding activities; training; determining requirements for protective equipment; conducting personnel monitoring; radiological surveys; and controlling radioactive shipments.

In addition, each individual whose occupational radiation dose is likely to exceed 100 mRem in a year will be instructed in their responsibilities for their own radiation safety and the controls required for general radiation safety.

12.5.2 Facilities, Equipment, and Instrumentation

12.5.2.1 Radiation Protection Facilities

12.5.2.1.1 Radiation Protection Laboratory

Radiation protection facilities at the Beaver Valley Power Station Unit 2 (BVPS-2) include a radiation protection laboratory with adequate equipment for detecting, analyzing, and measuring the types of radiation of concern, and for evaluating radiological problems which may be anticipated. Counting equipment including Geiger-Mueller, scintillation, and proportional counters and scalers, and equipment for evaluating samples is provided in an appropriately designed counting room for detecting and measuring radiation.

12.5.2.1.2 Decontamination Facility

A decontamination facility is provided at the station for decontamination of equipment and tools. Equipment and tools will be surveyed and decontaminated to the Department of Transportation Standards, as a minimum, prior to being shipped from the site.

12.5.2.1.3 Change Room Area

A change room has been provided in order that personnel may obtain protective clothing prior to entry into radiologically controlled areas of the station. These facilities are divided into clean and radiologically controlled sections. The radiologically controlled section is used to remove potentially contaminated clothing and is provided with sinks, showers, and necessary monitoring equipment to prevent the spread of radioactive contamination to the clean section. These areas are routinely surveyed for contamination, and decontamination of both the clean and radiologically controlled section is performed, when required.

12.5.2.1.4 Radiochemistry Laboratory

The radiochemistry laboratory (RCL) is designed and equipped for safe handling of unsealed radioactive sources. A separate chemistry laboratory is provided for all nonradioactive chemistry functions. Adequate laboratory equipment is provided for the two laboratories so that each is completely independent of the other. The wall, bench tops, shelving, and floor surfaces of the RCL are constructed of materials designed to facilitate decontamination.

RCL liquids are discharged to the radioactive liquid waste system. At least two (2) fume hoods are provided for handling and processing solutions. The hoods are maintained at a negative pressure and exhaust to a common monitored ventilation system vent through a filtering network.

Radiation monitors are provided in the RCL, both for general radiation surveillance of the area and for use as personnel monitors for personnel exiting the laboratory. A radioactive particulate air monitor is available for use when laboratory activities indicate the need for such additional monitoring. Personnel dosimetry is worn by personnel at all times while in the RCL. Protective equipment such as respirators, gloves, goggles, aprons, lab coats, coveralls, and shoe covers are available in the area to be worn as needed.

12.5.2.2 Radiation Protection Equipment and Instrumentation

12.5.2.2.1 Protective Clothing

Protective clothing is provided at BVPS-2 to protect personnel and to control the spread of radioactive contamination. Such clothing is designated as anti-contamination clothing (Anti-C).

Protective clothing will be worn as required by posted instructions in the work area and/or as directed by the Radiological Work Permit (RWP). The use of Anti-C clothing is restricted to radiologically controlled areas, unless other use is approved by radiation protection supervision. All persons will remove Anti-C clothing prior to exiting radiologically controlled areas of the plant.

Protective clothing available at the plant may include coveralls, underclothing, socks, shoe covers, rubber boots, plastic or rubber suits, goggles, face shields, cloth hoods, skull caps, cotton gloves, and rubber gloves.

Anti-C clothing is normally obtained at the entrance to the radiologically controlled area. Practices related to handling protective clothing will be detailed in written procedures.

12.5.2.2.2 Respiratory Protective Equipment

Respiratory protective equipment will be used to minimize the intake of radioactive material when engineering controls are not practicable, consistent with maintaining TEDE ALARA.

Some equipment protects against only a single type of hazard or a limited number of hazards, while other types provide protection against a broad class of hazards. The selection of respiratory equipment is made by personnel with appropriate knowledge of the requirements and limitations of such equipment. The proper respiratory device is selected to protect the wearer from unnecessary inhalation of airborne radioactive materials.

Respiratory protective devices used at BVPS-2 for radiological purposes will be the full-face design, sealed about the face and under the chin. Included may be the full-face mask equipped with high efficiency particulate filters, supplied air constant flow respirators with full face masks, supplied air hoods, and self-contained breathing apparatus.

A regular maintenance and inspection program for respirator equipment will be established.

12.5.2.2.3 Portable Equipment and Instrumentation

Portable instruments allow radiation protection personnel to perform alpha, beta, gamma, and neutron surveys in conjunction with area radiation monitoring, airborne radioactive particulate monitoring, and surface contamination monitoring.

Instruments used to perform such surveys shall be in current calibration and checked for source response each day they are used if appropriate. Equipment selected will provide the appropriate detection capabilities, ranges, sensitivities and accuracies required for the anticipated types and levels of radiation to be found at BVPS-2 to provide for all routine and anticipated emergency situations. The types of portable instruments, as well as the methods and procedures for use of portable instruments will be presented in written procedures.

12.5.2.2.4 Laboratory Equipment and Instrumentation

Equipment will be available to allow radiation protection personnel to perform laboratory evaluation of the radioactivity of samples. Determinations may be made of gamma, beta, and alpha emitting samples. The samples will be prepared and counted in appropriate counting geometries.

Calibration of laboratory equipment and instrumentation will be performed at regular intervals using appropriate sources traceable to the National Bureau of Standards/National Institute of Standards and Technology.

12.5.2.2.5 Personnel Monitoring Instruments

The personnel dosimetry system is designed to reliably evaluate radiation doses received at BVPS. It consists of TL dosimeters, direct reading dosimeters, and other components necessary to properly assess personnel dose.

The TLD is designed to be carried by personnel in the TLD badge holder. The TLDs will be assigned and issued to Radiation Workers and other individuals, as required by 10 CFR 20, who are expected to require access to a radiologically controlled area at BVPS. TLD data provides accurate measurement of radiation exposure, and will normally be the permanent record of external radiation dose at BVPS.

TLDs will be processed and the dose determined at intervals not to exceed the processor's recommendations, and may be processed more frequently as requested by radiation protection or station management. The TLDs will be response checked at a frequency specified by the processors National Voluntary Laboratory Accreditation Program (NVLAP).

Direct reading pocket dosimeters will be issued to each individual who requires access to any BVPS radiologically controlled area or to those designated by radiation protection supervision. The direct reading dosimeters are used for determining short-term accumulated exposures. Radiation protection staff will periodically record the direct reading dosimeter readings for purposes of determining TLD processing frequency, possible work restrictions, or other dose control measures. The direct reading dosimeters will be calibrated at intervals no greater than that specified by the dosimeter manufacturer.

Assessment of neutron dose equivalent will be based upon neutron survey measurements and exposure times or upon processed TLD results.

Personnel monitoring equipment will also include means of detecting personal contamination and may include the use of friskers, portal monitors, and body scanning equipment.

12.5.2.2.6 Other Radiation Protection Equipment

The Area Radiation Monitoring System (ARMS) consists of independent monitoring channels which, under normal conditions, monitor selected plant areas and warn personnel if radiation levels increase above prescribed levels. The area monitors utilize two basic types of detectors - ion chambers and Geiger-Mueller detectors. They provide indication both locally and in the Control Room. The monitors will be observed for proper operation, including proper response to check sources. They will be calibrated at regular intervals. Portable air monitors and area monitors may be used to supplement the system.

The Process Radiation Monitoring System consists of radiation monitoring channels for measuring the levels of radiation in various liquid and gaseous process streams. The process monitors use either beta or gamma scintillators as the basic detector and may be read directly in the Control Room. Each channel has alarms with settings selected to ensure radiation releases do not exceed prescribed levels. The Process Monitoring System will have regular instrument channel functional tests, as well as regular source checks. Calibration will also be performed regularly.

12.5.3 Procedures

Written procedures prescribe the policies, guidelines, and limits for all operations at BVPS which involve or potentially involve radiological consequences.

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

The radiological limits, controls, and policies as described in written procedures will be consistent and in compliance with the recommendations and requirements of applicable federal and state regulatory agencies. In accordance with both company and station administrative policies, procedural compliance is mandatory.

Radiation protection procedures will be supplemented by other work procedures to assure that abnormal or unique work practices and/or operations are conducted in accordance with the radiation protection program. This will ensure standards and limits specified therein are carefully and conservatively followed so that radiation exposures, contamination (surface or airborne) levels, and radioactive waste volumes are maintained as low as reasonably achievable and consistent with established requirements.

Written procedures will contain information on external radiation protection, respiratory protection, protective clothing, etc.; including detailed specifications, qualifications, precautions, performance limits, and/or controls and any other information, instructions, or guidelines required to minimize personnel exposure.

The radiation protection program has been developed in accordance with 10 CFR 20 and other applicable federal and state regulations. Quality assurance requirements related to radiation protection are implemented at BVPS-2 as described in Regulatory Guide 1.33. Revisions in applicable regulations that are adopted and published in the Federal Register, or are issued in the NRC Regulatory Guides will be evaluated. If warranted, radiation protection procedures will be revised to be consistent with the revised rules and practices.

The radiation protection program and its implementation will be audited periodically by qualified person(s), who are not assigned to the station, to ensure the program is in compliance with existing requirements. Audit findings will be reported to management for their information and appropriate action.

12.5.3.1 ALARA

Physical and administrative controls will be instituted to ensure the philosophy of maintaining personnel exposures (TEDE) are maintained as low as reasonably achievable (ALARA) is implemented.

12.5.3.1.1 Physical and Administrative Controls

Operation of the plant, as well as certain maintenance and repair tasks, may require access to and work in all areas of the plant. This includes areas defined by 10 CFR 20, as radiation areas and high radiation areas. Access time in higher radiation fields is limited and will be of short duration. Access time is based on the time required to perform the immediate task and the dose accumulation of the worker(s).

The following physical and administrative controls are standard operating practices used at BVPS-2 to minimize exposure to plant personnel:

1. Any area where potential radiological risks may be present is designated as a radiologically controlled area. In accordance with this practice, the primary plant and a portion of the service building are defined as the basic Radiologically Controlled Area. This includes the containment, the auxiliary building, the fuel building, the decontamination room, the PCA shop, certain concrete enclosed cubicles, and some adjacent service areas. Other radiologically controlled areas may be established outside the basic Radiologically Controlled Area, if warranted. Procedures are provided to govern access to and work within these areas.

2. A Radiological Work Permit (RWP) is required for any work within a radiologically controlled area. The permit will provide, as applicable, data on the radiological conditions which will be encountered, the authorized exposure limit for each member of the work party, and the specific radiological controls to be followed in accomplishing the task. The permit is used to assist in personnel exposure control. Implementation of the RWPs is detailed in written procedures.

3. Plant radiation areas, as defined by 10 CFR 20, will be maintained within established radiologically controlled areas. It is expected that areas in the plant where radiation levels may exceed one (1) mrem/hr will normally be within the basic Radiologically Controlled Area.

   Radiation areas are posted in accordance with 10 CFR 20. Written procedures define the specific posting and access requirements.

4. High radiation areas, as defined by 10 CFR 20, will be posted in accordance with 10 CFR 20, and access to high radiation areas and very high radiation areas will be controlled in accordance with Technical Specifications.

5. During reactor operation, high radiation areas may result in certain areas which otherwise would not be high radiation areas. Operational, radiological, and administrative controls are in place to preclude unauthorized entry to these areas. Entries into the Reactor Containment Building or the Volume Control Tank Area during reactor operations will require authorization of the Nuclear Shift Supervisor of the applicable unit. The procedures for authorization to enter these areas are contained in written procedures.

6. Radiation exposure to plant personnel is minimized to as low as reasonably achievable (ALARA) by providing specifically designed trenches for radioactive pipe, pipe tunnels, and shielded areas for radioactive piping and equipment. In addition, radiation exposure to plant personnel is limited by compliance with 10 CFR 20 and radiation protection procedures.

7. Dose control criteria are utilized to establish administrative control of the accumulated exposure of each individual. The purpose of this control mechanism is to ensure that all exposures are kept within 10 CFR 20 limits and ALARA. Dose control criteria are detailed in radiation protection procedures.

8. Standard operating procedures require that each individual working in a radiologically controlled area wear a direct reading pocket dosimeter and, normally, a TLD. The direct reading dosimeters should be read frequently by the individual to increase awareness of exposure. They are also read and recorded by radiation protection staff in accordance with procedural requirements. Direct reading dosimeters are normally used to supplement the TLDs, and they are checked in accordance with NRC Regulatory Guides 8.4 and 8.28. TLDs are read at specified intervals or as required by the dose control criteria. Normally, the TLDs are processed and the doses determined on a semi-annual basis.

9. Doses to members of the public will be maintained within the limits set forth in 10 CFR 20.

   Some examples of methods that will be used to maintain exposures ALARA are discussed for the following operations:

12.5.3.1.2 Refueling

Measures beyond those usually practiced for maintaining ALARA exposures will be taken during the refueling process. These include degassing and sampling the reactor coolant system to verify that the gaseous activity is minimized prior to removing the reactor vessel head. The reactor vessel head area will also be degassed prior to vessel head removal to avoid possible release of gaseous activity. The reactor is refueled with fuel handling equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an effective radiation shield. After flooding the reactor cavity and the refueling canal, purification of the water will be continued to maintain minimal radioactivity in the water, and therefore help keep radiation exposures ALARA.

12.5.3.1.3 Inservice Inspection

Radiation exposure for those personnel involved with Inservice Inspection will be minimized by observing basic ALARA considerations. These include prior planning of the work task in order to limit the amount of time spent in the radiation field, reducing the sources of radiation by processing, providing shielding, where possible, maintaining as much distance from the sources as possible, and following any other control methods specified by written procedures. In some instances, exposure rates may also be lowered by receiving regulatory relief from inspection requirements which subject personnel to excessively high radiation levels.

12.5.3.1.4 Radwaste Handling

The processing and handling of radioactive solid waste may involve high levels of radiation. Handling such waste must, therefore, be done efficiently and expeditiously in order to maintain the radiation exposures attributable to these operations ALARA. Exposures will be maintained ALARA by plant design. Liquids, spent resins, spent filter cartridges and compressible solid wastes will normally be processed remotely with the necessary shielding provided. Solidified waste shipping containers may also be remotely decontaminated, if necessary, prior to shipment off site. Shipping will be accomplished with the proper shielding provided.

12.5.3.1.5 Spent Fuel Handling, Loading, and Shipping

To keep personnel exposures ALARA, all movements of unshielded, spent fuel will be done under water utilizing fuel handling equipment. This provides shielding and cooling of the spent fuel. The water will be of sufficient depth to limit the radiation at the surface to approximately 5 mrem/hr during the time that the spent fuel assembly is being handled at its maximum allowable height within the refueling cavity.

The spent fuel pool is designed to hold spent fuel assemblies in underwater storage for a suitable decay period after the assemblies are removed from the reactor. After this period, the fuel may be removed from the racks and loaded into a shipping cask for eventual removal from the site. Prior to removal from the site, the cask may be cleaned and decontaminated.

12.5.3.1.6 Normal Operations

Control of airborne contaminants and gaseous radiation sources, radiation monitoring systems, and radiation shielding are features of BVPS-2 included in plant design to help keep exposures ALARA during normal operations.

The plant ventilation system provides filtered air and maintains areas of possible radioactive contamination at negative air pressure in order to ensure a safe and suitable environment for personnel and to limit the spread of airborne contaminants.

Radiation monitoring equipment is installed in selected areas of the plant to monitor and warn personnel of increasing radiation levels which might result in a radiation health hazard. Various process streams are also monitored for increasing radiation levels, which could indicate a possible plant malfunction and/or the existence of a radiation health hazard. Plant design provides proper shielding of equipment likely to contain or process large quantities of radioactive material.

12.5.3.1.7 Routine Maintenance

Radiological Work Permits (RWPs) help to ensure ALARA exposures during routine maintenance. RWPs are normally required for maintenance repair work within a radiologically controlled area. The RWP specifies the minimum requirements for anti-contamination clothing protective measures to be exercised and provides radiological conditions of the work site.

Where applicable, other appropriate steps will be taken to minimize exposure of personnel during the performance of their work functions. Prior planning of work tasks conducted in radiation-affected areas will be undertaken, including estimation of anticipated exposure, rehearsal of the planned task, planning of the lowest manpower requirements consistent with safe operation, and conducting as much work outside of the radiation area as possible. Installation of temporary shielding and the use of special tools will be considered for work tasks. Area decontamination may be performed to reduce both radiation levels and the probability of spread of contamination. Where practicable, flushing, filtering, draining, and other processing measures may be employed to reduce radiation levels.

12.5.3.1.8 Sampling

Periodic sampling of process streams will help keep radiation exposures ALARA, and provide confirmatory radioactivity information. Also sampling and monitoring of effluents from the waste disposal systems are performed to control effluent releases. During sampling, radiological hazards - contamination and radioactive airborne materials may exist. Necessary radiological controls will be observed, including avoiding contamination of the outside of the sample container and wearing the required Anti-C's during sampling operations. Radiological protection of personnel from sample system lines is provided by biological shielding. The sample panels and sinks will also be fitted with ventilation hoods to remove any radioactive gases released from samples.

12.5.3.1.9 Calibration

Periodic calibration of radiation detection instruments will help keep radiation exposures ALARA by assuring that the instruments are accurate and are providing reliable information. Instruments are maintained only by personnel who are properly trained to perform such maintenance.

Practices for maintaining exposures ALARA during source calibration include: ensuring proper operation of source calibrators and initiating required protective actions in the event of a malfunction of, or damage to, an instrument; providing proper interlocking systems used to prevent subjecting any personnel to an exposed source unnecessarily; limiting the time the source calibrator is in the exposed source position; and providing proper shielding for source storage.

12.5.3.2 Radiation and Contamination Surveying

12.5.3.2.1 Methods of Surveying

Specific requirements, such as those approved for radiation and contamination survey methods (including documentation and area posting) are described in detail in the radiation protection procedures. Survey results will be recorded on a "survey map" of the specific area, the radiological work permit, and/or other appropriate log.

The frequency and extent of radiation and contamination surveys will be determined in accordance with the criteria discussed in the following sections.

12.5.3.2.1.1 Radiation Surveys

Routine surveys of radiation fields will be performed, as described subsequently and are governed by written procedures.

1. Whenever operations are performed that can be expected to change radiation levels, radiation surveys will be conducted as necessary to control personnel exposure.

2. Boundaries of temporary radiation areas will be monitored periodically to ensure that radiation areas have not extended beyond the posted boundaries.

3. When highly radioactive equipment is moved within or into an area, gamma surveys will be made in adjoining work areas (including spaces above and below, if applicable).

4. Gamma surveys shall be performed as necessary to control radiation exposure in normally accessible areas which may require occupancy restrictions because of radiation, and accessible radioactive material storage areas. Surveys in long-term storage areas need to be conducted only upon personnel entry into the areas.

5. Potentially contaminated ducts, piping, and hoses outside the radiologically controlled area shall be surveyed for gamma radiation at a frequency commensurate with the potential for contamination as specified in written procedures.

6. Gamma radiation surveys will be performed in unlocked radiologically controlled areas where radioactive materials are not stored at a frequency commensurate with the potential for conditions to change, as specified in written procedures.

7. Surveys will be conducted when working with spent fuel handling containers, when removing shielding, and when opening shipping and storing containers.

8. Gamma radiation surveys will be performed on the outside of packages of radioactive material received on the site, and during unpacking of these containers. Shipping containers and packing materials should be surveyed prior to release for unrestricted handling.

9. Other surveys shall be performed, as necessary, to control personnel exposure to gamma, beta, neutron, and alpha radiation. Such surveys should include a gamma survey during initial entry into a tank connected to potentially radioactive piping, a gamma and neutron survey in spaces where significant radiation levels may exist from the operating reactor, and beta-gamma surveys of ventilation filters.

10. ALARA principles will be considered when determining survey frequencies.

12.5.3.2.1.2 Surface Contamination Surveys.

Routine surface contamination surveys will be performed with the frequencies indicated below, or more often, if necessary. Detailed procedures for conducting surface contamination surveys are provided.

1. Area contamination surveys will be performed within contaminated areas and at the exits from these areas at a frequency commensurate with the potential for changes in the contamination levels as specified in written procedures.

2. A contamination survey will be scheduled and performed in selected area(s) outside the radiologically controlled area. The survey area(s) will be varied to ensure all areas of the station are checked within a given period of time.

3. In addition, operations such as the following may require contamination surveys:

    a. Planned equipment decontamination may require surveys of the equipment being decontaminated.

    b. Areas where radioactive liquid leaks have occurred or where high airborne radioactivity levels exist may require additional surveys to determine the need for anti-contamination clothing and to determine the extent of the contamination area.

    c. Surveys will be taken prior to initial entry into tanks or voids containing potentially radioactive piping and when opening ventilation exhaust ducting.

4. ALARA principles will be considered when determining survey frequencies.

12.5.3.2.1.3 Airborne Radioactivity Samples

Detailed procedures for airborne particulate sampling are provided.

Airborne particulate samples shall be taken with portable air samplers:

1. At least every four hours in occupied areas when a potential airborne hazard exists. When practicable, continuous airborne radioactivity monitoring may be used in lieu of sampling.

2. When opening a radioactive system to the atmosphere for maintenance without total containment.

3. Before initially entering tanks connected to potentially radioactive piping.

4. Whenever airborne radioactivity concentrations are expected to exceed prescribed levels.

   Equipment, training, and procedures will be available for accurately determining the airborne radioiodine concentration in areas within the facility where plant personnel may be present during an accident.

   Portable instruments using selective sample media, such as charcoal or silver zeolite, will be used to provide effective monitoring of radioiodine levels. A sufficient number of portable samplers will be available to sample all vital areas of the plant.

   The capability to accurately measure the radioiodine concentrations present on the sample media, as well as the capability to remove the sample to a low-background, low- contamination area for further analysis, will exist.

12.5.3.3 Personnel Monitoring Systems

A comprehensive personnel monitoring system will be developed to monitor the radiation exposure received by persons working in or making visits to BVPS. This program will meet all requirements established by the Code of Federal Regulations, the Commonwealth of Pennsylvania, the Department of Health, and the Occupational Safety and Health Act of 1970.

External radiation exposure of plant personnel and construction workers is controlled for each individual in accordance with the dose control criteria detailed in radiation protection procedures. The purpose of the dose control criteria is to ensure that all exposures are maintained within 10 CFR 20 limits and ALARA. Each person is authorized to receive exposure based on these criteria. The individual is informed of this authorization and it is made available to his supervisors. The authorization is based on prior exposure data and station administrative guidance levels, or 10 CFR 20 limits.

An exposure tracking system provides the current radiation exposure status and/or exposure authorization for each individual. This information is factored into job assignments and assists in maintaining exposures ALARA.

In addition, access to high radiation areas is under the control of station supervision and/or radiation protection supervision. All persons who may have routine access to radiation areas (as defined in 10 CFR 20) must wear direct reading pocket dosimeters and TLDs when entering such areas. Direct reading pocket dosimeters are periodically checked and the TLDs are processed at regular intervals (or as required by the dose control criteria). Radiation levels in all accessible areas are maintained ALARA.

Personnel monitoring equipment consisting of direct reading dosimeters and/or dosimetry badges containing thermoluminescent dosimeters are assigned to and worn by personnel within the radiologically controlled areas of BVPS. In addition, those persons who work in radiation areas or contaminated areas may be required to wear extremity badges. The exact monitoring equipment used in a particular work situation will depend on the conditions encountered in the work area, and may consist of a combination of the preceding equipment, as necessary, to adequately monitor personnel exposure. In addition, portable air sampling equipment, including personnel air samples (lapel samplers) may be required for situations in which it is anticipated that airborne contamination may be encountered.

12.5.3.4 Radiation Control Procedures

Exposure to operating personnel is kept ALARA and maintained within the limits of 10 CFR 20. This is accomplished by careful design and administrative procedures and controls.

Radiation protection procedures describe radiological controls and practices at BVPS-2. This manual includes a number of measures which will be applied to work tasks in order to reduce personnel radiation exposure. These measures will be reviewed during the preparation of work procedures to determine if they are applicable to the work situation and/or environment. If the measures apply, the work situation will be evaluated to determine if significant exposure reduction will be accomplished and, if significant exposure reduction is likely, the applicable measures will be applied in order to attain the ALARA goal.

12.5.3.4.1 Prior Planning and Rehearsal

Prior planning of work tasks will be used to reduce personnel exposure to ionizing radiation. This planning will minimize the time spent in the radiation field, limit requirements for ingress to the work area, and allow better overall control. Rehearsals will be used for portions of particularly complex tasks to ensure the tasks are completed without delay. When practicable, tryout of containment devices or enclosures, trial fit of tools and parts, and similar measures will be used to reduce total personnel exposure.

12.5.3.4.2 Shielding and Decontamination

Temporary shielding and area decontamination will be utilized to reduce personnel exposure to ionizing radiation. Attention will be given to the installation of temporary shielding for work tasks where high radiation fields exist. Decontamination may also be used to reduce radiation levels in an area. Special beta shielding (eye protection, Anti-C clothing, etc.) shall be employed in work tasks involving high levels of beta radiation.

12.5.3.4.3 Special Tools

Consideration will be given to the use of special tools which may increase the speed of accomplishing the work task or provide distance between the worker and the radiation source. Special tools will be evaluated to determine if they are radiologically effective in reducing personnel exposure. Special tools will be used when specified by plant procedures.

12.5.3.4.4 Reduction Sources

When practicable, reduction in radiation levels will be accomplished by flushing systems, discharging systems, etc.

12.5.3.4.5 Radiological Work Permit and Surveys

Radiation surveys will be conducted prior to the start of work in radiation areas. The necessary radiological information will be identified on the Radiological Work Permit. Location of hot spots in the work area will be identified and marked. The permit will also specify the radiological exposure control methods which must be used during the performance of the specific assignment.

12.5.3.4.6 Work Practices

Work practices will be specified in radiation protection procedures and followed by all personnel working in radiologically controlled areas.

12.5.3.4.7 Radiological Training

Personnel who perform work in radiologically controlled areas will receive training in radiological control practices, including the proper use of protective clothing and respiratory protective devices. Additionally, personnel who perform work in radiation areas will be instructed in and directed to observe: the health protection problems associated with exposure to radiation and the precautions and procedures to minimize exposure; the protection of personnel from exposures to radiation; their responsibility to report promptly to the radiation protection organization any condition which may lead to or cause a violation of NRC regulations or unnecessary personnel exposure to radiation; and the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation. Personnel will be advised as to the radiation exposure reports which they may request pursuant to 10 CFR 19.13.

12.5.3.4.8 Job Debriefing

Debriefing of personnel will be conducted following jobs involving new or complex problems, and a critique prepared to improve future methods of performing the job. Additionally, feedback to improve procedures will be obtained through the job supervisor and periodic general safety meetings.

12.5.3.5 Dose Control

12.5.3.5.1 Internal Radiation Exposure

Consistent with maintaining TEDE ALARA, the criteria for control of internal radiation exposure are based on the aspect that airborne radioactivity should be minimized to such an extent that monitoring to determine internal exposure should not be necessary.

12.5.3.5.1.1 Respiratory Protection Program

The BVPS-2 respiratory protection program will not be used in place of practicable engineering controls and prudent radiation safety practices. Every reasonable effort will be exerted to prevent potential, and minimize existing airborne concentrations. However, when the potential for high airborne particulate activity exists, full-face high-efficiency filter masks, supplied air masks or hoods, self-contained breathing apparatus, or evacuation of the area (if no suitable protective devices are immediately available) may be exercised. Use of respiratory protection is based on ALARA considerations.

At a minimum, all respiratory protective devices used for radiological control purposes at the site are the full-face design. Included may be the full-face mask equipped with high efficiency particulate filters, supplied air constant flow respirators with full face masks, supplied air hoods, and self-contained breathing apparatus.

Personnel will be trained in the proper use of respiratory protective equipment. They will be cautioned as to the limitations of the respirators and instructed in methods to ensure a proper fit. Special ocular spectacles designed specifically for use with the particular respiratory equipment will be provided. The respirators will be handled and inspected in accordance with the guidelines provided in NRC Regulatory Guide 8.15 and NUREG-0041.

12.5.3.5.1.2 Internal Exposure Assessment/Bioassay Program

The internal dosimetry program, developed to meet the intent of Regulatory Guides 8.9 and 8.26, will identify those individual workers whose internal exposure may exceed applicable monitoring thresholds. It will require that a sufficient number of measurements on these individuals be performed so that an estimate can be made of their intakes. Normally, intakes are estimated without dose assessment unless the intake exceeds trigger level values.

The minimum internal dosimetry program is designed to give coverage to employees exposed to significant airborne radioactivity or contamination. For administrative purposes, all employees assigned to BVPS as qualified radiation workers shall be included in this program except those exempted as described in radiation protection procedures. Selected contractor personnel may also be included in the program.

Special consideration for selection for bioassay may be appropriate for those individuals who may have been exposed to abnormally high concentrations of radioactivity in the air, for individuals with previous internal exposures which require work restrictions, and for certain cases in which an individual is required to wear respiratory protection devices.

As warranted, the analysis or bioassay will consist of an in-vivo count of the lung and thyroid regions to determine the presence of high energy gamma emitters. In addition, in-vitro analysis or urinalysis will be required when significant quantities of non-gamma emitting nuclides are present in the working environment, should exposure conditions warrant such analyses.

The bioassay analysis may be performed once annually except where air sampling data exceeds certain criteria, in which case additional bioassays should be performed. Bioassay frequency is determined considering instrument sensitivity such that individuals whose internal exposure may exceed applicable monitoring thresholds will be identified. Bioassays will normally also be performed for persons prior to their assuming or terminating duties as a Radiation Worker.

12.5.3.5.2 Exposure Records and Reports

Administrative exposure guidance will be established and implemented to assure the limits of 10 CFR 20.101 are not exceeded and personnel occupational exposures are maintained ALARA. An occupational radiation exposure records system will be developed in accordance with NRC Regulatory Guide 8.7.

Occupational radiation exposure received during previous employment will be determined as required by 10 CFR 20. Records of prior dose will be maintained on NRC Form-4, or equivalent, and retained until the NRC authorizes disposition.

Records of occupational dose of individuals who are required to be monitored during employment or work assignment at BVPS will be maintained on NRC Form-5 or equivalent and retained until the NRC authorizes disposition.

Reports of exposure to radiation or radioactive materials will be made in accordance with 10 CFR 19, 10 CFR 20, and 10 CFR 50.

12.5.3.6 Radioactive Material Handling and Storage Methods

Methods of handling and storage of sealed and unsealed byproduct, source, and special nuclear material have been developed to provide control and accountability of the above material.

For byproduct material an accountability system, maintained by the radiation protection organization, will provide the following: documentation of all transfers to and from the site of byproduct material; identification of material and the necessary labels; inventory logs, maintained current by the radiation protection organization, to account for all radioactive sources; security precautions against unauthorized use of reference and calibration standards including locked storage areas for standards not in use; the conduct of a thorough search for any lost material and the notification of other site organizations, who will initiate appropriate action in the event of such loss.

For source and special nuclear material, a similar accountability system will provide for the following: documentation of receipt and shipment of nuclear material to and from the site, as well as transfers of material within the site; allowance of any special packaging, handling or shielding requirements of the material to be specified by the radiation protection organization; security precautions regarding nuclear material; specifically designated Item Control Areas (ICAs) for storage of nuclear material; control over movement of material from ICAs without authorization; the observance of proper criticality control techniques when nuclear material is moved; continual updating of book inventory, regular physical inventories and initiation of appropriate action in the event of the apparent loss of nuclear material.

Detailed procedures to implement the above accountability system are provided.