AMENDMENT 1 TO RESAR-SP/90 PDA MODULE 11, "RADIATION PROTECTION"

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INSTRUCTION SHEET

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471.1 In Module 11, subsection 12.2.1.1, Sources for Full Power Operation, second paragraph, item 4, the sentence: "Detailed angular distribution of radiation leakage (neutron and gamma ray) from the reactor pressure for streaming analysis" is not clear. Please clarify this statement.

Westinghouse Response:

Adding the word "vessel", which was inadvertently omitted, after the word "pressure" in line 4 should clarify this statement. See Attachment 4.7.1.

471.2 The subsection 12.2.1.1 Sources for Full Power Operation, the third paragraph refers to Table 12.2-1, which is not included in Module 11. Provide Table 12.2-1.

Westinghouse Response:

The referenced Table 12.2-1 (attached) was inadvertently omitted.

Note: A page of text immediately preceding Table 12.2-1 is also missing. The missing material is attached (Attachments 471.2a and 471.2b).

471.3 The acceptance criteria in Standard Review Plan Section 12.2 "Radiation Sources," (NUREG-0800), require that (for PWR's designed for recycling of tritiated water) the tritium concentrations in the contained sources and ~~airborne concentrations in the contained sources and~~ airborne concentrations in the regions specified in Item I.2 (of SRP 12.2) should be based on primary coolant concentration of 3.5 μCi/gm. State whether or not RESAR-SP/90 is in compliance with this requirement.

Westinghouse Response:

The design basis for tritium concentration in the primary coolant for the radiation sources presented in Section 12.2 is 3.5 μc/gm.

Note: Original NRC question has been changed to be consistent with SRP section 12.2.

471.4 State in subsection 12.2.1.3, Sources for Design Basis Accident, whether or not the shielding design for vital areas is based on radiation sources specified in NUREG-0737, Section II.B.2, and commit that the design dose rates for personnel in vital areas requiring continuous or infrequent access, will be such that the guidelines of GDC-19 will not be exceeded during the course of the accident.

Westinghouse Response:

A new paragraph should be inserted after the third paragraph in subsection 12.2.1.3 reading as follows:

> Shielding design for vital areas is based on radiation sources specified in NUREG-0737, Section II.B.2, and the design dose rates for personnel in vital areas requiring continuous or infrequent access, will be such that the guidelines of GDC-19 will not be exceeded during the course of the accident (see Attachment 471.4).

471.5 The last sentence in the second paragraph of subsection 12.2.1.3, Sources for Design Basis Accident, states that the integrated gamma ray and beta particle source strength for various time periods following the postulated accident are given in Table 12.2-19. Specify the location of the radiation source.

Westinghouse Response:

Although the location of the radiation source is implied in the first sentence of the paragraph, i.e., ... "released from the fuel to the containment ...", the last sentence in this paragraph should be modified to clarify the location. The last sentence should read as follows (see Attachment 471.4(a)):

> The integrated gamma ray and beta particle source strengths for various time periods, of fission products released from the fuel to the containment following the postulated accident, are given in Table 12.2-19.

Note: In Table 12.2-19, the exponents in the first five entries of the column headed "1 Year" are incorrect. The exponents 6,8,7,6, and 5 should be 14,14,13,13, and 13, respectively (Attachment 471.5).

471.6 State in subsection 12.3.1.1.1.2, Auxiliary Equipment, in Item D, whether or not pumps containing high level radiation sources are equipped with drain connections.

Westinghouse Response:

Under Item D, a sentence will be added which reads:

> For pumps containing high level radiation sources, means are provided for pump drainage prior to servicing.

See Attachment 471.6.

471.7 NUREG-0737, Section II.B.2, Shielding, specifies that areas requiring continuous occupancy should have less than 15 mrem/hr (average over 30 days). The control room and onsite technical support center are areas where continuous occupancy will be required.

Module 11, Subsection 12.3.1.2, Radiation Zoning and Access Control, describes such areas. However, the onsite technical support center is not included.

Include the onsite technical support center in the areas requiring post accident continuous occupancy.

Westinghouse Response:

The Technical Support Center (TSC) will be added to areas requiring post accident continuous occupancy (See Attachment 471.7).

471.8 Module 11, subsection 12.3.4.1, Area Radiation Monitoring, second paragraph states that the design of the fuel pool racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 70.24 and Regulatory guide 8.12 are not needed.

You are required to request an exception to 10 CFR 70.24 on your fuel storage license from the Office of Nuclear Material Safety and Safeguards; state in this subsection whether the exception has been granted.

Westinghouse Response:

A request for exception to 10 CFR 70.24 is the responsibility of the Utility Applicant. Westinghouse supplies the Utility Applicant with a criticality analysis report which provides justification.

471.9 State whether or not the Containment High Range Monitors A-11A and A-11B described in Section 12.3.4.1.8(H) are in compliance with NUREG-0737, Item II.F.1.3, Containment High Range Radiation Monitors.

Westinghouse Response:

These monitors are in compliance. Under Item H, a sentence will be added, which reads:

> "These monitors are in compliance with NUREG-0737, Item II.F.1.3".

(See Attachment 471.9).

471.10 In subsection 12.3.4.1.9, Range and Alarm Setpoints, in the third paragraph the statement regarding the "visual indication" of the area monitor reading in the control room is not clear.

State whether or not area monitors are equipped with strip chart recorders located in the control room.

Westinghouse Response:

The "visual indication" in the control room is a display on a cathode ray tube (CRT).

The area monitors are not equipped wtih strip chart recorders in the control room but the data is collected in a computer with printout on demand.

471.11 Subsection 12.4.1.1, Direct Radiation Dose Estimates, fourth paragraph states that in 1982 the highest and lowest collective dose was about 1600 and 100 respectively. Indicate the units for the collective dose levels.

Westinghouse Response:

The units are "man-rem".

On page 12.4-2, second paragraph, "man-rem" will be inserted after "100".

The first sentence in this paragraph will be modified to read:

> In 1982, the highest and lowest collective dose was about 1600 and 100 man-rem, respectively (Attachment 471.11).

471.12 You should take a position regarding the conformance to applicable NRC Regulatory Guides as specified in NUREG-0800, Standard Review Plan, in acceptance criteria of Section 12.1, 12.2, 12.3, 12.4 and 12.5. Each Regulatory Guide should be addressed separately. The applicant should state whether he complies with the Regulatory Guide in full or in part. The, exceptions, if taken, should also be specified and alternative methods for complying with the Commission's regulations should be described.

Westinghouse Response:

The positions regarding the conformance of RESAR SP/90 PDA to applicable NRC Regulatory Guides as specified in NUREG-0800 are as follows:

> The Regulatory Guides presented in the acceptance criteria of SRP Sections 12.1 and 12.5 are primarily for operations and therefore the responsibility of the Utility Applicant. However, RESAR SP/90 is in compliance with the design considerations of Regulatory Guide 8.8 which is included in these sections.
>
> For the applicable Regulatory Guides presented in the acceptance criteria of SRP Sections 12.2, 12.3 and 12.4, RESAR SP/90 complies or meets the intent of the guides as they relate to radiation protection design considerations. The one exception is Regulatory Guide 8.12, Criticality Accident Alarm Systems, (see response to NRC question 471.8).

471.13 Please provide following information required by TMI Action Plan Item II.B.2 on page II.B.2-5 of NUREG-0737, "Clarification of TMI Action Plan Requirements":

(1) Specification of source terms used in the evaluation; including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation. If any of the systems listed in "Clarification," item 2, were excluded, explain why such systems are excluded from review;

(3) Specification of areas where access is considered necessary for vital system operation after an accident (partially answered, refer to Question 471.7). If any of the areas listed in the "Clarification" section of II.B.2 were not considered to be areas requiring access after an accident, explain why they were excluded;

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

Westinghouse Response:

The above four items, which are given on page II.B.2-5 of NUREG-0737, are prefaced on that page by "... For operating license applicants ..." and therefore is not required for a PDA.

471.14 The compliance with NUREG-0737, Section III.D.3.3, Improved Inplant Iodine Instrumentation under Accident Condition, requires that a low background counting laboratory is available. Please address this item.

Westinghouse Response:

The low background counting laboratory is the responsibility of the plant specific applicant. Additionally, section III.D.3.3 of NUREG-0737, under "Applicability", states "This requirement applies to all operating reactors and all applicants for an operating license." and therefore is not required for a PDA.

Note: Westinghouse is supplying a change to page 12.2-1 of Module 11 to maintain consistency with RESAR-SP/90 PDA Module 12, "Waste Management", Attachment 471.W1 reflects that change.

3. Gamma ray dose rates at the inside surface of the primary shield concrete.

4. Detailed angular distributions of radiation leakage (neutron and gamma ray) from the reactor pressure vessel for streaming analyses.

The nitrogen-16 activity of the coolant (produced from oxygen activation) is the controlling radiation source in the design of the secondary shield and is tabulated in Table 12.2-1, in microcuries per gram of coolant, as a function of transport time in a reactor coolant loop. The nitrogen-16 source in the pressurizer is given in Table 12.2-2.

Fission and corrosion product activities circulating in the reactor coolant and out-of-core crud deposits comprise the remaining significant radiation sources during full power operation. The fission and corrosion product activities circulating in the reactor coolant are given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management". The fission and corrosion product source strengths in the reactor coolant pressurizer liquid and vapor phases are given in Table 12.2-2. The isotopic composition and specific activity of typical out-of-core crud deposits are given in Table 12.2-3. Typically, 1 milligram of deposited crud material is found in one square centimeter of a relatively smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high-affinity-material areas, and possibly thin-boundary-layer regions.

Systems which process or contain reactor coolant also contain radiation sources during full power operation. These systems include the chemical and volume control system (CVCS) and the boron recycle system (BRS), as described in RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Table 12.2-4 gives the radiation sources for the CVCS, specifically delineating the sources for:

1. CVCS letdown stream.
2. Mixed bed demineralizers.
3. Cation bed demineralizer.

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in $\mu Ci/cm^3$ in the applicable region

From the above equation, it is evident that the peak or equilibrium concentration, C_{Eqi}, of the i^{th} radioisotope in the applicable region will be given by the following expression:

$$C_{Eqi} = (LR)_i \; A_i \; (PF)_i / V\lambda_{Ti}$$

With high exhaust rates, this peak concentration will be reached within a few hours.

12.2.3 REFERENCES

1. Lutz, R. J., and Chubb, W., "Iodine Spiking - Cause and Effect," ANS Transactions, Vol. 28, pg. 649, June, 1978.

2. DiNunno, J. J., et al, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March, 1962.

ATTACHMENT 471.2(b)

TABLE 12.2-1

RADIATION SOURCES

REACTOR COOLANT NITROGEN-16 ACTIVITY

Position in Loop	Loop Transit Time (sec)	Nitrogen-16 Activity (microcuries/gm)
Leaving core	0.0	163
Leaving reactor vessel	3.3	118
Entering steam generator	3.8	112
Leaving steam generator	8.9	68
Entering reactor coolant pump	9.6	64
Entering reactor vessel	10.7	57
Entering core	12.3	53
Leaving Core	13.2	163

Nitrogen-16 Energy Emission

Energy (Mev/gamma)	Intensity (Percent)
1.75	0.13
2.74	0.76
6.13	69.0
7.12	5.0

irradiation period of 400 days. Irradiated incore flux thimble gamma ray source strengths are given in Table 12.2-18. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per cubic centimeter of Inconel-600 for an irradiation period of 15 years. The flux thimbles are made of Inconel-600 with a maximum cobalt impurity content of 0.10 weight percent.

12.2.1.3 Sources for Design Basis Accident

The radiation sources of importance for the design basis accident are the containment source and the residual heat removal system source.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in TID-14844 (Reference 2). These assumptions are consistent with those provided in Regulatory Guide 1.4 and Section II.B.2 of NUREG-0737. The integrated gamma ray and beta particle source strengths for various time periods, of fission products released from the fuel to the containment following the postulated accident, are given in Table 12.2-19.

The post-accident recirculation system and shielding should be designed to allow limited access to the high head safety injection (HHSI) and the residual heat removal pumps following a maximum credible accident. The sources are based on the assumptions in TID-14844 with only the nongaseous activity being retained in the sump water, which flows in the residual heat removal loop. Noble gases formed by the decay of halogens in the sump water are assumed to be released to the containment and not retained in the water. Gamma ray source strengths for radiation sources circulating in the residual heat removal loop and associated equipment are given in Table 12.2-20.

Shielding design for vital areas is based on radiation sources specified in NUREG-0737, Section II.B.2, and the design dose rates for personnel in vital areas requiring continuous or infrequent access, will be such that the guidelines of GDC-19 will not be exceeded during the course of the accident.

ATTACHMENT 471.4(b)

Isotopic fission product sources from the maximum credible accident, based on the assumptions in TID-14844, are given in Chapter 15 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System".

12.2.2 Airborne Radioactive Material Sources

Sources of airborne radioactive material in equipment cubicles, corridors, or operating areas normally occupied by operating personnel from systems and components described in RESAR-SP/90 may be obtained from the reactor coolant activities given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management".

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance, etc.) and operating procedures followed during and after shutdown. The permissible coolant activity levels following depressurization should be based on the noble gases evolved from the reactor coolant system water upon removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification should be established based on maximum permissible concentration considerations and containment ventilation system capabilities of the plant. Operating plant experience has indicated that coolant xenon-133 concentrations of less than 0.05 microcuries per gram have posed no problem to the containment atmosphere during vessel head removal.

The exposure rates at the surface of the reactor cavity and spent fuel pool water are dependent on the purification capabilities of the reactor vessel cavity and spent fuel pool cleanup systems. A water activity level of less than 0.005 microcuries per gram for the dominant gamma emitting isotopes at the time of refueling has been shown in operating experience to maintain the dose rate at the water surface to less than 2.5 millirem per hour.

12.2.2.1 Model for Calculating Airborne Concentrations

For those regions which are characterized by a constant leakrate of the radioactive source at constant source strength and a constant exhaust rate of

ATTACHMENT 471.5

TABLE 12.2-19

INTEGRATED GAMMA RAY AND BETA SOURCE STRENGTHS AT VARIOUS TIMES FOLLOWING A MAXIMUM CREDIBLE ACCIDENT (TID-14844 Release Fractions)

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt) 0.5 Hour	1 Hour	2 Hours	8 Hours	1 Day
0.2 - 0.4	1.3×10^{12}	2.3×10^{12}	4.0×10^{12}	1.3×10^{13}	3.3×10^{13}
0.4 - 0.9	8.4×10^{12}	1.5×10^{13}	2.4×10^{13}	5.3×10^{13}	8.6×10^{13}
0.9 - 1.35	3.7×10^{12}	6.4×10^{12}	1.1×10^{13}	2.5×10^{13}	3.8×10^{13}
1.35 - 1.8	3.6×10^{12}	6.3×10^{12}	1.0×10^{13}	2.1×10^{13}	3.0×10^{13}
1.8 - 2.2	1.9×10^{12}	3.2×10^{12}	5.2×10^{12}	1.1×10^{13}	1.3×10^{13}
2.2 - 2.6	2.0×10^{12}	3.7×10^{12}	6.1×10^{12}	1.2×10^{13}	1.4×10^{13}
2.6 - 3.0	3.4×10^{11}	5.5×10^{11}	8.4×10^{11}	1.3×10^{12}	1.4×10^{12}
3.0 - 4.0	3.6×10^{11}	4.6×10^{11}	5.8×10^{11}	8.0×10^{11}	8.3×10^{11}
4.0 - 5.0	1.6×10^{11}	1.6×10^{11}	1.7×10^{11}	2.0×10^{11}	2.0×10^{11}
5.0 - 6.0	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}
Beta	1.3×10^{13}	2.2×10^{13}	3.6×10^{13}	8.9×10^{13}	1.5×10^{14}

	1 Week	1 Month	6 Months	1 Year
0.2 - 0.4	1.3×10^{14}	2.3×10^{14}	2.6×10^{14}	2.6×10^{14}
0.4 - 0.9	1.7×10^{14}	2.7×10^{14}	5.3×10^{14}	6.4×10^{14}
0.9 - 1.35	4.8×10^{13}	5.3×10^{13}	6.0×10^{13}	6.4×10^{13}
1.35 - 1.8	4.6×10^{13}	7.2×10^{13}	8.6×10^{13}	8.9×10^{13}
1.8 - 2.2	1.4×10^{13}	1.5×10^{13}	1.8×10^{13}	2.0×10^{13}
2.2 - 2.6	1.5×10^{13}	1.7×10^{13}	1.8×10^{13}	1.8×10^{13}
2.6 - 3.0	1.4×10^{12}	1.5×10^{12}	1.5×10^{12}	1.5×10^{12}
3.0 - 4.0	8.4×10^{11}	8.5×10^{11}	8.5×10^{11}	8.5×10^{11}
4.0 - 5.0	2.0×10^{11}	2.0×10^{11}	2.0×10^{11}	2.0×10^{11}
5.0 - 6.0	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}	1.1×10^{10}
Beta	3.9×10^{14}	6.5×10^{14}	1.1×10^{15}	1.4×10^{15}

evaporator components are separated from those that are less radioactive. Instruments and controls are located in accessible low background radiation areas.

D. Pumps

For pumps containing high level radiation sources, means are provided for pump drainage prior to servicing. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease of removal.

E. Tanks

In general, horizontal and flat-bottom tanks are sloped downward to the tank drain. Overflow lines are directed to the waste collection system to control any contamination within plant structures. For tanks outside structures, which can potentially contain radioactive fluids, dikes are used to contain overflows.

F. Heat Exchangers

Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials to minimize leakage. Impact baffles are provided, and tube side and shell side velocities are limited to minimize erosive effects. Wherever possible, the radioactive fluid passes through the tube side of the heat exchanger.

G. Instruments

Instrument devices are located in low radiation zones and away from radiation sources whenever practical. Primary instrument devices, which for functional reasons are located in high radiation zones are designed for easy removal to a lower radiation zone for calibration.

to component crud traps or radiation streaming, but design features are incorporated to minimize such effects and the higher dose rates are expected to be highly localized and/or intermittent. Actual in-plant zones and control of personnel access will be based upon surveys conducted by the plant health physics staff.

Areas which may require occupancy to permit an operator to aid in the long term recovery from an accident are considered in the design. Such areas include the control room, Technical Support Center (TSC), safety-related motor control centers and switchgear, post accident sampling system room, radiochemistry laboratory, and remote shutdown panels. Such radiation protection design features are described in Section 12.3.2 of this module. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates in the area, and appropriate time limits for presence in the area are imposed.

Ingress or egress of plant operating personnel to controlled access areas is controlled by the plant health physics staff to ensure that radiation levels and exposures are within the limits prescribed in 10 CFR 20. Any area having a radiation level that could cause a whole body exposure in any 1 hour in excess of 5 mrem, or in any 5 consecutive days in excess of 100 mrem, will be posted "Caution, Radiation Area." Radiation areas are provided with access alert barriers, e.g., chain, rope, door, etc. Any area having a radiation level that could cause whole body exposure in any 1 hour in excess of 100 mrem will be posted "Caution, High Radiation Area." High radiation areas (> 100 mrem/hr) are provided with locked or alarmed barriers. During periods when access to a high radiation area is required, positive control is exercised over each individual entry. To the extent practicable, the measured radiation level and the location of the source is posted at the entry to any radiation area or high radiation area.

The posting of radiation signs, control of personnel access, and use of alarms and locks are in compliance with requirements of 10 CFR 20.203.

E. Sampling Room Area Monitor A-6

To continuously indicate the radiation levels in the sampling room. A high radiation alarm signal warns the occupants of the sampling room of a deteriorated radiological condition.

F. Seal Table Instrumentation Room Area Monitor A-7

To continuously indicate the radiation levels in the seal table room and establish radiological habitability prior to entry. A high radiation alarm signal warns occupants of the seal table room of a deteriorated radiological condition.

G. Containment Access Hatch Area Monitor A-9

To continuously indicate the radiation levels in the containment access hatch and establish radiological habitability prior to entry.

H. Containment High Range Area Monitors A-11A and A-11B

To indicate, along with A-2A and A-2B, the radiation levels inside the containment building at the operating deck following a design basis accident. A high alarm signal initiates containment isolation phase A to mitigate the consequences of a design basis accident (primarily a LOCA). These monitors are in compliance with NUREG-0737, Item II.F.1.3.

12.3.4.1.9 Range and Alarm Setpoints

The range, setpoints, and control function of the PERMS area monitors are given in Table 12.3-3. The setpoints are initial and are subject to modification as plant operating experience is developed.

Radiation zones are described in Table 12.3-1.

- o In general, dips or valleys in the curve were due to new plant start-ups. For example, the 1973-1975 time period had twelve new plants come on-line. The low doses at these plants during their first year of operation lowered the average plant dose.

- o In 1982, the highest and lowest collective dose was about 1600 and 100 man-rem, respectively. In addition, the average collective dose dropped to 650 man-rem.

Factors which may have contributed to this increase in plant collective doses include the following[2]:

- o Increasing plant radiation fields
- o Required or mandated modifications/back-fits
- o Premature failures of major components
- o Use of inexperienced workers
- o Management attitude

In the analysis of ORE data cumulative man-rem per cumulative MW_e-Yr of electricity generated accounts for plant size, and provides a relative measure of costs (man-rem) versus benefits (power production). Table 12.4-1 provides a cumulative summary (up through 1982) of collective doses for domestic plants with Westinghouse-supplied NSSSs. Also included on Table 12.4-1 is a measure of the effective operating time (MW_e-Yr/MW_e).

As it can be seen from Table 12.4-1, cumulative man-rem per MW_e-Yr ranges from 0.30 to 2.68. The best performing plants operate in the range of 0.3 to 0.4 man-rem per MW_e-Yr. Major factors which have contributed to these excellent performance levels include low plant radiation fields, good layout and access provisions, and excellent operational practices and procedures. If

12.2 RADIATION SOURCES

This section discusses and identifies the sources of radiation that form the basis for shield design calculations and the sources of airborne radioactivity used for the design of personnel protection measures and dose assessment.

12.2.1 Contained Sources

The shielding design source terms are based upon the three plant conditions of normal full power operation, shutdown, and design basis accident events.

12.2.1.1 Sources for Full Power Operation

The primary sources of radioactivity during normal full power operation are direct core radiation, coolant activation processes, leakage of fission products from pinhole defects in fuel rod cladding, and activation of reactor coolant corrosion products. The design basis for the shielding source terms for fission products in this section is cladding defects in fuel rods producing 0.25 percent of the core thermal power. The design basis for activation and corrosion product activities is derived from measurements at operating plants and is independent of fuel defect level. The radionuclide activity levels in the reactor coolant based on a realistic (or expected) model are given in Section 11.1 of RESAR-SP/90 PDA Module 12, "Waste Management", as are the models and assumptions used in determining these sources.

Westinghouse provides, to the applicant, numerous reactor radiation source values for the at-power condition, including:

1. Neutron particle fluxes at the inside surface of the primary shield concrete at the core midplane.

2. Gamma ray energy fluxes at the inside surface of the primary shield concrete at the core midplane.