APPENDIX 5-1

ISSUE SUMMARY
Form SOP-0402-03, Revision 2

<table>
<tr><td colspan="3">DESIGN CONTROL SUMMARY</td></tr>
<tr><td>CLIENT: BNFL
PROJECT NAME: Big Rock Point Major Component Removal
PROJECT NO.: 10525-020
CALC. NO.: N-10525-020-0002
TITLE: Transport Package Shielding Design
EQUIPMENT NO.: N/A</td><td>UNIT NO.: N/A
☐ NUCLEAR SAFETY-RELATED
☐ NOT NUCLEAR SAFETY-RELATED
☒ IMPORTANT TO SAFETY- CATEGORY A</td><td rowspan="4">QA SERIAL NO.</td></tr>
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Initial Issue: 43 pages in calculation body & 28 pages total in Attachments A through M.
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☒ VERIFIED
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PREPARER G.D. Cole /A. G. Klazura — DATE:
REVIEWER * R. Kahn — DATE:
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REVIEWER* — DATE:
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* The reviewer's signature indicates compliance with S&L procedure SOP-0402 and the verification of, as a minimum, the following items: correctness of mathematics for manual calculations, appropriateness of input data, appropriateness of assumptions, and appropriateness of the calculation method.

NOTE: PRINT AND SIGN IN THE SIGNATURE AREAS

Rev. Date: 01-29-1999
Eff. Date: 02-15-1999

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1.0 Purpose and Scope

1.1 Purpose

The purpose of this calculation is to determine upper bound shielding compositions and thicknesses that comprise the Big Rock Point's Reactor Vessel Transport System (RVTS) package. The integrated unit to be packaged is designated the Reactor Vessel Assembly and Internals (RVAI). The RVAI itself is radioactive and contains all radioactive components that will, once packaged, become the radioactive shipment.

This revision to the calculation incorporates the following changes:

- revise the reactor vessel wall source term and add the grid bar end pieces source term to the RVAI and demonstrate that the top, bottom, and side thicknesses of the Big Rock Point RVTS package are sufficient to maintain the contact and 2 meter dose rates within acceptance criteria limits;
- evaluate the reactor vessel nozzle penetrations in the vicinity of the grid bar end pieces for radiation streaming through these penetrations;
- determine the unshielded dose rate at 1 meter from the top, bottom, and side of the reactor vessel;
- recalculate dose rates attributed to radiation source terms in the reactor vessel wall; and
- demonstrate that the configuration and design of the plugs used to seal low density cellular concrete (LDCC) fill holes in the RVTS system package do not have a significant impact on calculated dose rates.

This revision was initiated because the eighteen grid bars (and associated radiation source terms) were not completely removed from the RVAI. The grid bars provide lateral alignment for the top of the fuel assemblies. The majority of the length of each grid bar was cut and was removed however, grid bar end pieces of 6" and 10" lengths were left in the reactor vessel. The radiation source term associated with the grid bar end pieces is significant and must be accommodated when addressing the shielding requirements for the RVTS. The reactor vessel wall is composed of carbon steel. The reactor vessel wall radiation source term was revised to reflect a cobalt-60 inventory representative of carbon steel. In the previous revision to this calculation, the cobalt content was conservatively based on a stainless steel reactor vessel wall material. Stainless steel has a higher cobalt content than carbon steel. With the inclusion of the grid bar end pieces source term, the additional conservatism associated with use of a stainless steel cobalt-60 source term in the reactor vessel wall was removed.

The methodology used in the previous revision to this calculation employed both the MicroShield [Reference 2.7] and the QAD [Reference 2.8] computer codes to determine dose rates around the RVTS. The dose rate evaluation using the MicroShield computer code was removed from this calculation. The QAD computer code can model complex geometries and can therefore determine dose rates around the RVTS more accurately. The MicroShield runs were originally included as an independent verification of the QAD model.

This revision to the calculation completely supercedes the previous revision.

1.2 Scope

This calculation determines reactor vessel transport container shielding thicknesses and compositions required to meet dose rate criteria and it evaluates the impact that reactor vessel nozzle penetrations and LDCC fill hole plugs (in the transport container wall) have on dose rates.

The output of this evaluation is calculated dose rates at contact to the surface of and 2 meters from the RVTS package. Additionally, this calculation determines unshielded dose rates at 1 meter and 3 meters from the RVAI.

1.3 Background

The RV Internals are already fixed in position, but a low-density cellular concrete (LDCC) will be pumped into the RV to surround the remaining internals. This operation serves to prevent the movement of superficial radioactive contaminants. The RV Head will not be a part of the RV integrated unit. Some other enclosure, like a steel plate, will be attached to the remaining RV Assembly and Internals in place of the RV Head.

The RVAI unit will be packaged for the purposes of transport to and near surface burial at the Barnwell, South Carolina Waste Management Facility. The packaged RVAI is the essential item that is a part of the RVTS. To create the RVTS package, the RVAI unit will be positioned in the package and surrounded by 50 lbs/ft^3 cellular concrete. The package itself will provide additional radiological shielding and be of sufficient construction to withstand the expected rigors of handling, transport and burial.

Federal regulations with regard to low-level radioactive waste, especially in Chapter 10 of the Code of Federal Regulations, Parts 20, 61 and 71, are accommodated in this calculation. The dose-rate limits specified in Part 71 receive primary consideration in this calculation with respect to the RVTS package's design.

2.0 References

2.1 "Reactor Vessel Heat Rates and Shielding", Calculation No. N-10525-020-0001, Big Rock Point Major Component Removal, Revision 1;

2.2 "Big Rock Point Reactor Vessel and Internals Characterization and Classification", Report WMG-9902, Revision 1, June 1999, WMG Project 8057;

2.3 "Revised Characterization and Classification Results; WMG Project 8057", Revision to Reference 2.1, March 10, 2000, WMG Inc.;

2.4 "Trojan Nuclear Plant Reactor Vessel and Internals Removal Project Safety Analysis Report", PGE-1076, Revision 1, July 1999, Portland General Electric Company;

2.5 Deleted;

2.6 Deleted;

2.7 Program MicroShield, Program Number 03.7.508-4.10, March 10, 1995, Sargent & Lundy LLC ;

2.8 Program ISOSHLD-PC, Program 03.7.310-1.0/O, QAD options, June 1, 1999, Sargent & Lundy LLC;

2.9 "RVTS Component Weights", DIT No. DIT-BNFL-MCR-001-01, S. Reed--Preparer, Issue Date 5/05/2000, Sargent & Lundy LLC;

2.10 "Handbook of Chemistry and Physics", Edited by D.R. Lide et al., 78th Edition, 1997-98, CRC Press LLC;

2.11 Code of Federal Regulations, Title 10, Volume 2, Part 61, Revised January 1, 1999, U.S. Government Printing Office via GPO Access, "Waste Classification" (...61.55), "(8) *Determination of concentrations in wastes;"*

2.12 "Nuclear Power Plants No Longer in Service", Nuclear News, March 2000, Vol. 43, Number 3, pg. 57, American Nuclear Society Inc., LaGrange Park, IL, USA;

2.13 Drawings: Consumers Power, Big Rock Point

2.13.1 Drawing 197E853, "Vessel & Core Arrangement," Revision 2

2.13.2 Drawing 794E830, "Steam Baffle," Revision 1

2.13.3 Drawing 767E939, "Core Spray Sparger," Revision 2

2.13.4 Drawing 794E839, "Steam Baffle & Core Spray," Revision 1

2.13.5 Drawing 198E118, "Top Plate," Revision 2

2.13.6 Drawing 237E956, "Thermal Shield Retainer," Revision 1

2.13.7 Drawing E-201-809-3, "Core Support Plate," Revision 3

2.13.8 Drawing M2-695, "Inlet Baffle," Revision A

2.13.9 Drawing F-230-791-2, "General Arrangement, Reactor Vessel," Revision 2

2.13.10 Drawing 198E179, "Reactor Vessel Insulation," 1/13/61

2.13.11 Drawing 104R175, "Assembly Reactor Vessel," Revision 6;

2.13.12 Drawing 197E861, "Top Guide," Revision 4;

2.13.13 Drawing SD-10525-020-001, "RVTS Cask", Revision A

2.14 "Nuclides and Isotopes," 14th Edition, General Electric Company;

2.15 NUREG/CR-1413, ORNL/NUREG-70, "A Radionuclide Decay Data Base – Index and Summary Table," D. C. Kocher, May, 1980;

2.16 "Radiation Protection by Shielding, Tables for the Calculation of Gamma Radiation Shielding" Paul-Friedrich Sauermann, 1976;

2.17 M. D. Papp (BNFL) to B. Slimp (S&L), "Big Rock Point Restoration Project - 5339 - Reactor Vessel Grid Bars and Neutron Windows," Letter No. BRP-2000-05-228 dated May 25, 2000

2.18 WMG Report WMG-20024-20091, "Concentration Averaging of Grid Bar End Pieces for Disposal of BRP Reactor Vessel and Internals," Prepared for BNFL, October 2000;

2.19 "Reactor Vessel Transport System Radiation Source Term," Calculation No. N-10525-020-004, Revision 1

3.0 Design Input

3.1 Units [Reference 2.10, pgs. 1-24 – 1-31, 1-38]:

a. 1 Curie (Ci) = 3.7 E+10 Bequerels = 3.7 E+10 disintegrations (dis.) /second;

b. 1 hour (hr) = 3.600 E+3 seconds (sec) ;

c. 1 pound (lb) = 0.4536 kilograms (kg) = 453.6 grams (g) ;

d. 1 inch (in) = 0.0254 meter (m) = 2.54 centimeter (cm) ;

e. 1 liter (L) = 1000 milliliter = 1000 cm^3 ;

3.2 Reactor Component Activities [Attachment A of Reference 2.3, Included as Attachment A to this calculation]:
Total radioactive inventories for RVAI source term components are obtained from the reference and have already been decayed to a date of September 1, 2002. The Neutron Windows have not been included since they will be removed from the RVAI before processing, packaging and shipment. Total activities for the RVAI components are presented in Table 3-1 that follows. Component activities are from Reference 2.3. The radionuclide distribution and radionuclide inventories in the reactor vessel wall are presented in Table 3-1.1 and are from Table 6.3.2-2 of Reference 2-19.

Section 1.0 and Table 1-1 of Reference 2.2 indicate specifically that the Top Guide ("grid bars") will not be part of the RVAI due to this component's classification as Greater-Than-Class C (GTCC). Reference 2.17 confirms that the Top Guide ("grid bars") and the Neutron Windows will be removed from the RV prior to packaging and shipment. A copy of Reference 2.17 is included as Attachment L to this Calculation. As indicated in Reference 2.18, grid bar end pieces, ranging in length from 10 inches (3 pieces) to 6 inches (15 pieces), were left in the reactor vessel. These grid bar end pieces are attached to the top guide plate. The rest of the grid bar lengths were cut and removed from the reactor vessel. The radiation source term associated with the grid bar end pieces that remain in the reactor vessel is 1800 Curies [Reference 2.19].

Table 3-1. Total Reactor Component Activities

Component	Activity (Curies)
Steam Baffle	1.08 E+1
Sparger	8.60 E -1
Top Guide Plate	1.14 E+3
Seal Housing	1.30 E+2
Thermal Shield (TS)	7.40 E+3
TS Retainer	1.14 E+2
Seal Weights	1.10 E+3
RV Insulation	5.81 E+1
RV Wall	1.11 E+3
RV Wall Cladding	2.45 E+2
Core Support Plate	1.40 E -1
Inlet Diffuser	2.24 E -2
Inlet Baffle	1.63 E -1
Grid Bar End Pieces	1.80 E+3
Total	1.31E+4

Table 3-1.1. RV Wall Inventory

Nuclide	Inventory (Curies)
H-3	3.62E-1
C-14	1.21E-1
Sb-125	2.23E-4
Mn-54	1.19E+0
Eu-152	1.40E-2
Fe-55	7.90E+2
Co-60	2.31E+2
Ni-59	5.27E-1
Ni-63	8.68E+1
Nb-94	1.82E-3
Tc-99	3.90E-4
Total	1.11E+3

3.3 Trojan Reactor Vessel Package Radioactivity Fractions [Reference 2.19 & Reference 2.4, Table 1-2; see Attachment B, "Activity, Activation (Curies)"].

Table 3-2. Trojan RV Radionuclide Curies

Nuclide	Activation (Curies)	Fraction*
H-3	6.55 E+2	3.26 E –4
C-14	2.18 E+2	1.08 E –4
Sb-125	4.04 E –1	2.01 E –7
Mn-54	2.16 E+3	1.07 E –3
Eu-152	2.54 E+1	1.26 E –5
Fe-55	6.97 E+5	3.47 E –1
Co-60	1.15 E+6	5.72 E –1
Ni-59	9.53 E+2	4.74 E –4
Ni-63	1.57 E+5	7.81 E –2
Nb-94	3.29 E+0	1.64 E –6
Tc-99	7.06 E –1	3.51 E –7
Total	2.01 E+6	1.0

*Note: The fraction values for some of the nuclides given in this table vary slightly in the 3rd decimal place from values presented in Reference 2.19. This difference is attributed to using the denominator value of 2.01E+6 when determining the nuclide specific fractions provided in this Table instead of using the value of 2.008E+6 which is obtained by summing the activation values over all nuclides.

3.4 PR-2 (Radiation Monitor) Accuracy; see Attachment C.

Accuracy: +/- 15% for all (monitor) ranges from 20% to 80% of full scale, calibrated with Cs-137.

3.5 The following vessel internal components are those residing nearest the top, measured axially, of the Reactor Vessel. See Reference 2.2, Table 2-1 and Figure 2-1.

Table 3-3. Top Section Components' Characteristics

Component	Thickness	Outer Diameter
Steam Baffle	1.0 in., grid pattern	105.50 in.
Sparger	2 in. Schedule 40 pipe	105.00 in.

The steam baffle consists of 4 plates attached via hinges to a circular ring [Drawing 197E853, Reference 2.13]. It extends from 20'3.25" to 20'4.25" above the base line. The base line is just above the bottom inside surface of the reactor vessel [Drawing 794E830, Reference 2.13].

The sparger is located directly below the steam baffle [Drawings 767E939 and 794E839, Reference 2.13].

3.6 The following vessel internal components are those residing in the region just above, measured axially, the "active region" components. The physical measurements for the top guide plate and the seal housing are taken from Reference 2.2, Table 2-1. Dimensions for the grid bar end pieces are obtained from Reference 2.18.

Table 3-4. Plate-Seal Irradiated Components' Characteristics

Component	Thickness	Outer Diameter
Top Guide Plate	1.0 in. wide	99.188 in.
Seal Housing	3.0 in. high	105.69 in.
Grid Bar End Pieces (6" & 10" long)	7/8 in. thick	5-5/8 in high

The top guide plate is a 1" thick annular ring with an irregular cutout in the center portion [Drawing 198E118, Reference 2.13.5]. The bottom of the top guide plate is 14'2.75" above the base line and 3" above the top of the active fuel region.

The thermal shield seal housing is a 3" high annular ring located at the top of the thermal shield between the thermal shield and the reactor vessel wall [Section 4.2 of Reference 2.2].

There are 18 grid bars and thus 18 grid bar end pieces. The grid bars are secured (via a pivot assembly) to the top guide plate at one end of the bar. The grid bars can pivot up into the vertical position or can pivot into the horizontal position. Each grid bar is a rectangular section that is 5-5/8" high and 7/8" thick. Three (3) grid bar end pieces are 10" long and 15 end pieces are 6" long [Reference 2.18]. The 10" long grid bar end pieces are located around 0° azimuth which is located in the central region of the arc of grid bar end pieces. The grid bar end pieces are secured to the top of the top guide plate (near the guide plate inner radius) and are evenly spaced from 270 degrees to 90 degrees azimuths [Reference 2.13.12]. When in the horizontal position, the bottom of the top row of grid bars is nominally 1.75" above the top of the top guide plate and the distance between the top of the top guide plate and the center of the pivot point is 2.625" [Reference 2.13.12]. When in the horizontal position, the grid bar end pieces extend about 0.75 inches horizontally (toward the reactor vessel wall) from the center of the pivot point as measured off Reference 2.13.12.

3.7 The active region components are those located along the elevation occupied by the reactor fuel which the maximal axial neutron flux impinged [see Reference 2.2, Figure 2-3] and that will remain to comprise the RVAI's predominant activity. The characteristics are taken from References 2.2 and 2.13.

Table 3-5. Active Region Irradiated Components' Characteristics

Component	Inner Radius	Outer Radius
Thermal Shield (TS)	50.00 in.	51.50 in.
TS Retainer	(rectangular	brackets)
Seal Weights	51.50 in.	52.50 in.
RV Wall	53.22 in.	58.47 in.
RV Insulation	58.47 in.	61.47 in.

The reactor vessel wall is 5.25" thick carbon steel. A 5/32" thick stainless steel cladding lines the inside of the reactor vessel wall. [Reference 2.2 and Drawing F-230-791-2, Reference 2.13].

The active fuel region is 70" in length. The bottom of the active fuel region is 8' 1.75" above the base line [Drawing 197E853, Reference 2.13]. The top of the reactor vessel flange is 282" above the base line [Drawing 104R175, Reference 2.13].

The thermal shield is 92" long. It extends 12" above and 10" below the active fuel region. The inner diameter of the thermal shield is 100" (i.e., 50" radius) and the outer diameter to the thermal shield external surface is 103" (i.e., 51.5" radius) [Drawing 197E853, Reference 2.13].

There are 12 thermal shield seal weights. The seal weights are located between the thermal shield and the reactor vessel wall. Each seal weight is composed of 3 plates, each plate is 1" by 18" wide by 27" long [Reference 2.2]. The seal weights extend from the bottom of the thermal shield to 4" above the active core [Drawing 104R175, Reference 2.13].

There are 6 thermal shield retainers. Per Drawing 237E956, Reference 2.13, the retainers extend from 5" below the thermal shield to 25.125" up from the bottom of the thermal shield. The retainers are about 10" wide at the top and 5" wide at the bottom half. The thickness of the retainer plate is about 0.75" as measured off the drawing and verified by information provided in Table 2-1 of Reference 2.2.

The reactor vessel insulation is 3" thick. It is composed of stainless steel plate with folded stainless steel foil between the plates [Section 4.2.3 of Reference 2.2]. The insulation extends 3" from the reactor vessel surface [Drawing 198E179, Reference 2.13].

3.8 The following vessel internal components are those residing nearest the bottom, measured axially, of the Reactor Vessel. See Reference 2.2, Table 2-1 and Figure 2-1.

Table 3-6. Bottom Section Components' Characteristics

Component	Thickness	Outer Diameter
Core Support Plate	1.50 in.	79.25 in.
Inlet Diffuser	5.50 in., ring	
Inlet Baffle	36.0 in., height	

The core support plate is a 71" by 71" plate with rounded corners and many penetrating holes [Drawing E-201-809-3, Reference 2.13]. The top of the core support plate is 13.5" above the base line [Drawing 197E853, Reference 2.13].

The inlet baffle is a 36" high, 5/8" thick hexagonal parallel piped hollow skirt steel plate [Table 2-1 and Section 4.2.2 of Reference 2.2]. The inlet baffle is located along the circumference of the reactor vessel. The bottom of the inlet baffle is just above the core support plate i.e., the bottom of the inlet baffle is a little greater than 18.5" above the base line [Drawing M2-695, Reference 2.13].

There are 2 inlet diffusers. Each diffuser is 27" long by 5.5" wide, by 3/8" thick [Table 2-1 of Reference 2.2]. The diffusers are located in front of i.e., on the reactor vessel side of the 20" diameter inlet nozzles [Drawing F-230-791-2, Reference 2.13].

3.9 Thermal Neutron Fluxes

Axial Fluxes

The following thermal neutron flux values were obtained via visual examination of the axial thermal flux distribution presented in Figure 2-3 of Reference 2.2.

Table 3-7. Axial Thermal Neutron Fluxes

Location	Neutron Flux Range (n/cm^2–sec)
From bottom of active core to 4" below	0.5 times active core average flux
From 4" to 15" below active core i.e., to bottom of thermal shield (TS) retainer	2E+12 to 8E+11
From bottom of TS retainer to 20" below retainer	8E+11 to 2E+9
For distances greater than 20" below TS retainer	2E+9 to 1E+5
From top of active core to top of TS	1.5E+13 to 5E+12
From top of TS to 20" above TS	5E+12 to 2E+11
From 20" above TS to steam baffle	2E+11 to 2E+9
Above steam baffle	2E+9 to 5E+7
Active fuel region	1.5E+13

The annulus region between the RVAI and the RVTS steel shielding will be filled with LDCC with a density range between 50 and 60 lbs/ft^3 [References 2.9 and 2.13.13]. Within the RVAI, the density of the cellular concrete employed will range from 30 and 36 lbs/ft^3. This less dense concrete will also span the baffle-sparger component layer near the top of the RV up to the RVTS package's top plate. The minimum LDCC densities of 50 lbs/ft^3 (in the annular region) and 30 lbs/ft^3 (within the RVAI) are used in this evaluation. Use of the minimum LDCC densities will maximize the calculated dose rates since gamma dose rates are inversely proportional to the shielding material density.

3.11 Densities for the various RV components will be determined from physical data included in Table 2-1 of Reference 2.2 (Table 2-3 of Reference 2.18 for grid bar end pieces). Those data are repeated in Table 3-8, as follows:

Table 3-8. RVAI Components' Physical Characteristics

Component	Weight	Volume
Thermal Shield (TS)	1.30 E+4 lbs	2.59 E+1 ft^3
TS Retainer	5.47 E+2 lbs	1.09 E+0 ft^3
Seal Weights	5.08 E+3 lbs	1.02 E+1 ft^3
RV Wall	2.03 E+5 lbs	4.14 E+2 ft^3
RV Insulation	5.18 E+3 lbs	2.59 E+2 ft^3
Core Support Plate	2.54E+3 lbs	5.08 ft^3
Grid Bar End Pieces (18 end pieces)	1.71 E+2 lbs	---

3.12 Reactor Vessel Nozzles:

Reactor vessel nozzles are identified on Drawing 198E179 [Reference 2.13.10]. The nozzles are associated with penetrations through the reactor vessel wall. The largest penetrations are the recirculating water inlet penetrations located near the bottom of the reactor vessel and the steam outlet penetrations located near the reactor vessel flange elevation. There are 2 nozzles for recirculating water inlet (each 20" in diameter) and 6 steam outlet nozzles (each 14" in diameter). The 20" diameter nozzles are angled downward 20 degrees from the reactor vessel bottom tangent line. The 14" diameter nozzles are 234" above the base line (i.e., 48" below the vessel flange). Penetrations N-12 and N-14 are located above and closest to the active fuel region of the reactor vessel. These penetrations are 3" in diameter, and are located at 108° and 252° azimuths respectively. The penetration centerlines are at 192" above the base line. There are other penetrating nozzles in the reactor vessel. The dose rates, due to radiation shine through these other penetrations, will be bounded by the dose rates through the 14" diameter and 3" diameter nozzles near the upper portion of the reactor vessel and by the dose rates through the 20" diameter nozzles near the bottom of the reactor vessel.

3.13 The reduction factor for 1.3 MeV gamma radiation attenuation through 5.25" (13.3 cm) of iron is 30. The factor of 30 is obtained from Table 31a (Iron) of Reference 2.16. The energy value of 1.3 MeV is related to Co-60.

3.14 Ni-63 and C-14 are beta emitters with average beta energies of 0.171 and 0.0495 MeV per decay, respectively [Reference 2.15]. Ni-59 and Fe-55 are low energy gamma emitters with average gamma energies of 0.0024 and 0.0017 MeV per decay [Reference 2.15].

3.15 The holes used to fill LDCC into the annular region between the reactor vessel and the transport container are shown in Reference 2.13.13. These holes are indicated in the notes to Reference 2.13.13 to be 3" in diameter with a tolerance of plus or minus 1/8". There are 4 LDCC fill holes vertically aligned along the outside annular surface of the transport container. Two fill holes are centered 3' and 6' from the top surface of the transport container. The other two LDCC fill holes are centered at 3'-6.5" and 8'-2.6875" from the bottom surface of the transport container. Metal plugs will be used to seal these holes. The plugs are 3" long with a ½" deep rectangular cutout on the surface to accommodate a screwdriver. They will be threaded into place and will be inserted 0.5" below the transport container shield wall surface. There are also 10 holes each with a diameter of 1-9/16" located in the top of the transport container which are used to insert LDCC into the reactor vessel. These holes are to be filled with threaded plugs having a diameter of 1-1/2" and a depth of 3". [Reference 2.13.13].

4.0 Assumptions

4.1 The relative radionuclide abundances from the Trojan RV activation activities are assumed to provide a more reliable and complete radionuclide profile for the Big Rock RVAI. This assumption is compelling for the following reasons:

a. The Trojan RV packaging, shipment, and burial, as well as the supporting Safety Analysis Report [Reference 2.4] were complete with full regulatory compliance in 1999;

b. While the June 1999 WMG Report [Reference 2.2] appears generally thorough, tritium (H-3) has a characterization that includes the designation, "<LLD>". It would be advantageous to be able to manifest a H-3 activity, especially, without the LLD designation. Employing the Trojan profile allows a basis by which to do this;

c. This calculation applies Trojan's profile to all of the RVAI components as an integrated unit (with some adjustment for the RV wall) because the RV Wall and Insulation components are not as fully characterized as all the other components in the WMG Report;

d. There is reasonable consistency between the Big Rock RVAI (0.53 from Attachment A; "LLRW" Corrected Co-60 to the Corrected Total Activities ratio) and the Trojan RV (0.57 from Table 3-2) when comparing Co-60 activation abundances; utilizing the Trojan profile effectively conservatively skews the Big Rock RVAI to greater Co-60 Curies;

e. The radionuclide decay time from the end of reactor operations to that for the activity determinations is the same for Big Rock's characterization (August 1997 to September 2002; see Attachment A) as for Trojan's (November 1992 to November 1997; Attachment B, Note 1). See Reference 2.12 for the August 1997 and November 1992 dates ("closed" dates).

This approach depends on the concepts given in Reference 2.11. The scaling implied above relies on the Trojan RV relative radionuclide abundances as meeting the regulation's intent with respect to "indirect methods [which] can be correlated to actual measurements". Application of the Trojan RV activation nuclide distribution to the Big Rock Point RVAI activation activities results in a conservative determination of shield thickness requirements for the RVTS package (i.e., calculated shield thickness requirements are maximized). Required shield thicknesses are conservatively calculated using the Trojan nuclide distribution because application of the Trojan distribution results in a greater overall inventory of Co-60 which is the dominant radiation source for shielding.

4.2 The annulus region between the RVAI and the RVTS steel shielding will be filled with LDCC (minimum density of 50 lbs/ft^3). Within the RVAI, the LDCC employed will have a minimum density of 30 lbs/ft^3. This less dense cellular concrete will also span the baffle-sparger component layer near the top of the RV up to the RVTS package's top plate. Since gamma dose rates are inversely proportional to the shield material density, use of LDCC with greater densities than those specified (i.e., 30 & 50 lbs/ft^3) will result in dose rates smaller than those calculated in this evaluation. The range of LDCC densities within the RVAI and in the annular region between the RVAI and the RVTS steel shielding is discussed in Design Input 3.10.

4.3 Material Compositions and Densities:
All reactor vessel components, except for the reactor vessel wall, are assumed to be composed of 304 stainless steel. The reactor vessel wall material is carbon steel. These material designations are obtained from Section 2.1.2 of Reference 2.2. The composition of 304 stainless steel and carbon steel is assumed to be that presented in Table 2-2 of Reference 2.2 which are taken from NUREG CR-3474. The compositions follow.

Table 4-1. Material Composition Data (Weight Percent)

Element	Material Type	
	304 SS	Carbon Steel
Nitrogen	0.045	0.008
Chromium	18.400	0.000
Manganese	1.530	1.350
Iron	70.600	97.570
Nickel	10.000	0.610
Molybdenum	0.260	0.580
Niobium	0.009	0.002
Cobalt	0.141	0.012

The shield material for the transport package is assumed to be iron with a density equal to that of pure iron i.e., 7.86 g/cc [Periodic Table of the Elements, Page 16 of Reference 2.14]. A density of 7.86 g/cc is assumed for carbon steel and 304 stainless steel.

An air density of 1.293E-3 g/cc is assumed [Reference 2.10].

The core support plate has many holes (Design Input 3.8). The density of the core support plate is assumed to be 0.5 times the density of iron.

The bottom of the reactor vessel also includes a number of holes. The density for the bottom of the reactor vessel is assumed to be 80% that of iron.

The ISOSHLD-PC and QAD evaluations use elemental compositions for stainless steel components with an assumed density of 7.86 g/cc. Solid component densities range from 7.86 g/cc to 8.04 g/cc based on component weight and volume information provided in Table 3-8 of Design Input 3.11. Use of the lower density maximizes the dose rates and is thus conservative.

4.4 Radiation Source Term Distribution
It is assumed that radionuclide activity is homogeneously distributed within the component or component portion under consideration.

When determining the radionuclide concentration in components that are at the elevation of the active fuel region and include elevations above and/or below the active fuel, it is conservatively assumed that all of the component activity identified in Table 3-1 is within the active fuel region of the component. Components which fall into this category are the reactor vessel wall/cladding,

thermal shield, thermal shield seal weights, thermal shield retainers, and reactor vessel insulation. It is further assumed that activities in portions of the components located above or below the active fuel region are directly proportional to the axial thermal neutron flux distribution.

There are 18 grid bar end pieces located near the inner radius of the reactor vessel wall (Design Input 3.6). When determining the dose rate due to radiation source terms in the grid bar end pieces, it is assumed that the radiation source term is homogenized in a 0 to 180 degrees annular region which represents the grid bar end pieces. This annular region is assumed to extend from the inner to outer radii of the top guide plate (i.e., from 44" to 50"). The vertical range of the zone designated in the QAD computer code for the grid bar end pieces extends from the top surface of the top guide plate to 12" above the top guide plate (i.e., about 4" greater than the value of 7.875" calculated in Section 6.9 of this calculation to accommodate the 10" long grid bar end pieces). The radiation source region within this QAD zone extends from about 2" above the top guide plate to just below 12" above the top guide plate. The dimensions for the annular region take advantage of existing QAD model boundaries and account for the grid bar end pieces whether in the horizontal or vertical positions.

4.5 Transport Package Overall Dimensions

It is assumed that the shielded transport package has an outside diameter of 13 feet, that the shield top of the transport package is bolted onto the reactor vessel flange at the head area, and that the minimum distance between the bottom of the reactor vessel outer surface and the inside surface of the transport package bottom shield is 4".

The thickness of the transport container shielding is assumed to be:

- Top: 4" thick iron
- Bottom: 3" thick iron
- Side: 7" thick iron from 16" above the top of the active fuel elevation to 10" below the bottom of the active fuel elevation and 3" thick everywhere else.

4.6 The steam baffle is assumed to cover 80% of the reactor vessel internal cross sectional area based on observation of Drawing 794E830 [Reference 2.13].

4.7 When quantifying the radiation source terms in components located near the bottom of the reactor vessel, it is assumed that the radiation source terms include 1 curie due to radioactive surface contamination having the same radionuclide distribution as that used for activation source terms. The activity due to surface contamination is added to the component activation activities as described in Section 5.2.3 to this calculation. The basis for 1 curie of surface contamination is Reference 2.19.

5.0 Methodology and Acceptance Criteria

5.1 Acceptance Criteria

The acceptance criteria must generally account for the dose rate limits applicable to the RVTS package (10 CFR 71.47), such that:

a. External (contact) dose rates shall not exceed 200 millirem/hour on any accessible surface;

b. In the case of an open transport vehicle, the dose rates shall not exceed 200 millirem/hour at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the package, and on the lower external surface of the vehicle;

c. At any point two meters from, in the case of an open vehicle, the vertical planes projected from the outer edges of the conveyance, 10 millirem/hour must not be exceeded.

5.2 Methodology for QAD Evaluation

The QAD option of the ISOSHLD-PC computer program [Reference 2.8] is used to determine dose rates for specified dose points situated around the shielded transport package. Contact dose rates at the surface of the transport package and dose points 2 meters from the transport package are considered. The QAD option implements the ISOSHLD-PC point and line kernel integration technique with the QAD complex source and geometry features to determine gamma dose rates at specified dose points. Gamma radiation from radionuclide decay and from electron Bremsstrahlung are considered. Energy dependent linear attenuation coefficients and buildup factors are derived from material specifications and code databases. The ISOSHLD-PC computer code (including the QAD option of ISOSHLD-PC) has been qualified under S&L's Quality Assurance program.

The QAD evaluation is used to define shield thickness requirements for the RV transport cask and to demonstrate that dose rates at all dose points around the shielded transport package are within acceptance criteria values after accounting for dose rate contributions from all the radiation sources and accounting for differences in shield thickness along the vertical side of the transport package. In addition to determining shielded dose rates at contact to and 2 meters from the shielded transport package, the QAD evaluation determines unshielded dose rates. Unshielded dose rates are determined at 1 meter and 3 meters from the side of the reactor vessel insulation along the midplane, at 1 meter and 3 meters from the top of the reactor vessel along the centerline, and at 1 meter and 3 meters from the bottom of the vessel along the centerline. When determining the unshielded dose rates, credit is taken for radiation attenuation due to 30 pounds per cubic foot cellular concrete in the reactor vessel. For the QAD computer model representing the unshielded case, air replaces the 50 pounds per cubic foot cellular concrete located between the reactor vessel and the transport container and air replaces the transport container iron material.

When determining the unshielded dose rates above the RVTS, buildup is determined based on a concrete material for radiation source terms associated with the thermal shield segments at and

above the active core midplane, source terms associated with the top guide plate, the thermal shield shield housing, the sparger, the steam baffle, and the grid bar end pieces. These source terms are located near the top of the reactor vessel and there are no significant quantities of iron shielding between these source terms and the dose points. Buildup determination using concrete as the buildup medium instead of iron results in slightly larger dose rates and is therefore conservative. Iron is used as the buildup medium for all other source term components for the unshielded evaluation above the RVTS, for RVTS side and bottom unshielded evaluations, and for the RVTS shielded evaluations.

The QAD evaluation determines dose rates for the following reactor vessel (RV) transport package shielding considerations:

- The top of the RV shielded transport package is a 4" thick iron plate attached at the reactor vessel flange elevation.
- The bottom of the RV shielded transport package is 3" thick iron.
- The side shield for the RV transport package is 7" thick iron from 10" below the bottom of the active fuel elevation to 16" above the top of the active fuel elevation and 3" thick iron elsewhere.
- A cellular concrete with a density of 30 pounds per cubic foot is used to fill all void spaces in the reactor vessel.
- A cellular concrete with a density of 50 pounds per cubic foot is used to fill all void spaces between the reactor vessel wall and the shielded transport package.

Input and Output files for the QAD model are identified in Attachment I to this calculation.

5.2.1 QAD Model

The model used in the QAD evaluation is shown in Figure H-1 of Attachment H. It consists of 41 zones and 34 boundaries. (Only 33 of the 34 boundaries are used in the final QAD runs. Boundary #33 was used in intermediate QAD runs to define a transitional iron thickness for RVTS annular side shield segments between 3 and 7 inches thick. Boundary 33 is the inner boundary for zones 38 and 39 that are located above and below the active core region.) A zone represents a component or space between or around components. The boundaries provide the dimension limitations for the zones. The relative locations of the components with respect to each other and the boundary dimensions were determined using the information given in Design Inputs 3.5 through 3.8. The boundary dimensions are shown in Figure H-1. Boundary dimensions are given in inches and are either radial lengths or elevations in the 'Z' direction. The origin for the coordinate system is at the midplane of the active fuel, along the active fuel centerline. Boundary numbers are identified in Figure H-1 with the letter 'B' followed by the boundary number. Zones are identified by a number enclosed within a box. A description of the zones is also provided in Attachment H.

The QAD model used to determine dose rates associated with the grid bar end pieces deviates slightly from the model presented in Figure H-1. A description of these deviations follows:

- Zones 38 and 39 (depicted in Figure H-1) were combined into zone 8.
- Zones 40 and 41 (depicted in Figure H-1) were combined into zone 7.
- Boundary 34 was renamed boundary 33 (since boundary 33 was not used as indicated above).

- A new zone 38 was created to represent the grid bar annular region. This zone extends from the top guide plate inner radius to its outer radius (i.e., from boundaries 23 to 1). And it extends from the top of the top guide plate (boundary 24) to the 7" to 3" RVTS annular shield transition (boundary 33).
- A new zone 39 was created to define the region between zone 38 and zone 27. This zone is an annular region of 30 lbs/ft^3 low density cellular concrete which extends radially from zone 38 to zone 27 (boundaries 1 and 2) and vertically from zone 2 to zone 41 (boundaries 25 and 26).
- A new zone 40 was created to define the region above zone 26 and adjacent to zone 38. Zone 40 is a disk of 30 lbs/ft^3 low density cellular concrete that extends radially to zone 38 (boundary 23) and vertically from zone 26 to the top of zone 38 (i.e., between boundaries 26 and 33).
- A new zone 41 was created to define the radial region between zones 38 and 4 (i.e., between boundaries 1 and 3) and the vertical region located above zones 39 and 27 and below zone 28 (i.e., bounded by boundaries 26 and 33). Zone 41 consists of 30 lbs/ft^3 low density cellular concrete.
- In addition to the deletion of zones 38 through 41 and the creation of new zones 38 through 41, zone/boundary interfaces in the vicinity of zones 7, 8, and 38 through 41 had to be redefined in the QAD code input.

The boundaries for the QAD zones which represent the RVAI radiation components are generally defined by the physical boundaries of the component. This is not the case for the QAD zone which represents the grid bar end pieces. The grid bar end pieces are homogenized in a semi-annular zone which includes the grid bar end pieces and in-vessel LDCC located between the grid bar end pieces. This annular zone has a radial thickness of 6" and a height of 12". The zone boundaries were chosen to provide a reasonable radiation source zone while making use of existing boundaries already defined in the QAD model. The annular zone thickness of 6" encompasses the grid bar width of 5-5/8" when the grid bars end pieces are in the vertical position and the annular thickness of 6" accommodates the 6" length of the majority of grid bar end pieces when in the horizontal position. The 12" height accommodates the maximum length of 10" for grid bar end pieces and it allows 2" for the distance between the top guide plate and the bottom of the grid bar end pieces.

5.2.2 Materials and Compositions Used in QAD Model

As indicated in Assumption 4.3, all of the components associated with the reactor vessel are 304 stainless steel except for the reactor vessel wall which is carbon steel. The densities for stainless steel, carbon steel, and iron are all assumed to be 7.86 g/cc (Assumption 4.3). The core support plate and reactor vessel bottom have a number of penetrations. The densities for these components were reduced to account for these penetrations. The density for the reactor vessel insulation is determined using the weight and volume information from Design Input 3.11. Components composed of carbon steel or stainless steel use the elemental weight percents given in Assumption 4.3 and the component density to determine the material composition densities. The ISOSHLD-PC/QAD iron material designator (ID# 26) is used for the transport container shield. The cellular concrete is modeled as ordinary concrete using the ISOSHLD-PC/QAD material designator (ID# 103) with the specified densities (i.e., 0.8009 g/cc for out-of-vessel 50 pounds per cubic foot cellular concrete and 0.4806 g/cc for in-vessel 30 pounds per cubic foot cellular concrete). Air is modeled using the ISOSHLD-PC/QAD air material designator (ID# 101) and a density of 1.293E-3 g/cc (Assumption 4.3). Elemental partial densities for

components composed of carbon steel and 304 stainless steel are presented in the table that follows. These densities are calculated in Section 6.3.

Table 5-1. Materials and Compositions Used in QAD Model

Material Composition Components	Shield Material Partial Densities (Grams/cc)					
	QAD Material ID#	304 SS	Carbon Steel	Vessel Insulation	Core Support Plate	Carbon Steel on RV Bottom
Nitrogen	7	3.537E-03	6.288E-04	1.442E-04	1.769E-03	5.030E-04
Chromium	24	1.446E+00	0.000E+00	5.895E-02	7.231E-01	0.000E+00
Manganese	25	1.203E-01	1.061E-01	4.902E-03	6.013E-02	8.489E-02
Iron	26	5.549E+00	7.669E+00	2.262E-01	2.775E+00	6.135E+00
Nickel	28	7.860E-01	4.795E-02	3.204E-02	3.930E-01	3.836E-02
Molybdenum	42	2.044E-02	4.559E-02	8.330E-04	1.022E-02	3.647E-02
Niobium	41	7.074E-04	1.572E-04	2.883E-05	3.537E-04	1.258E-04
Cobalt	27	1.108E-02	9.432E-04	4.517E-04	5.541E-03	7.546E-04

When determining unshielded dose rates associated with the grid bar end pieces source term, the grid bar region (i.e., zone 38) is modeled as air with a density of 1.293E-3 grams per cubic centimeter. This will maximize calculated unshielded dose rates since self attenuation in the source term region of air is insignificant. When determining the shielded dose rates associated with the grid bar end pieces source term, the density for the grid bar end pieces region was modeled as the homogenized iron density of the grid bar end pieces and the in-vessel LDCC. The homogenized density was treated as concrete in the QAD model. This homogenized density is calculated in Section 6.3 of this calculation.

5.2.3 Radiation Source Term Activity Used in QAD Model

The dose rate to each dose point was determined separately for each radiation source term component. The dose rates from all the source term components were added to obtain the total dose rate. Many of the components are of annular geometry which encompass an angle from 0 to 360 degrees. When determining shielded dose rates, the annular source region was limited to the region between 30 and 150 degrees i.e., 1/3 of the full circumference. A scale factor of 1/3 was included in the code input to maintain the proper activity concentrations. Using the reduced source region reduced the run time requirements and code instabilities associated with the complex geometry. Dose rate results at contact and 2 meters for the limited regional source model were compared to the full 360 degree source model for a number of runs of different source regions and noted differences were insignificant (less than 1%). The dose points at the side of the transport package were located at 90 degrees. This places the dose points centrally with respect to the radiation source. Calculated dose rates at the top and bottom centerlines of the transport container are multiplied by 3 to account for the total annular source.

The grid bar end pieces are distributed fairly evenly over an arc of 180 degrees i.e., their location ranges from 270 to 90 degrees (Design Input 3.6). The grid bar end pieces source region is modeled as an annular homogenized source region from 0 to 180 degrees. For the QAD shielding evaluation, the annular source region was limited between 30 and 150 degrees which

reduced the run times and code instabilities encountered when the full source range was used. The entire grid bar end pieces source activity was homogenized within this annular segment. This results in conservative calculation of shielded side dose rates because the source concentration is larger than actual by about a third. The 18 grid bar end pieces are distributed fairly evenly along the semicircle. Use of an homogenized source term in an annular source region is acceptable due to the distribution of the grid bar end pieces and the relative distances between the source and dose points. As indicated in Section 5.2.1 of this calculation, the QAD zone for the grid bar end pieces is a 6" wide by 12" high annular region. In the QAD model, the grid bar end pieces source term is distributed along the entire 6" width of the zone and from 1.875" to 11.875" (i.e., 10") within the 12" zone height. This 10" source height represents the maximum length of the grid bar end piece. The value of 1.875" represents the distance between the top guide plate and the grid bar end piece. [When in the horizontal position the grid bar end pieces are a maximum of 1.75" + 5-5/8" = 7.375" above the top guide plate (Design Input # 3.6). When in the vertical position, the grid bar end pieces would extend from ((2.625" – 0.75") to (10" + 2.625" – 0.75")) i.e., 1.875" to 11.875" above the top guide plate. The distances of 0.75" and 2.625" are identified in Design Input # 3.6. The value of 10" is the maximum length of the grid bar end pieces.]

The component activities given in Table 3-1 and the Trojan nuclide distribution given in Table 3-1.1 (for the reactor vessel wall) and Table 3-2 (for other RVIA components) are used in the QAD evaluation. Component activities from Table 3-1 are increased by 15% to account for radiation monitor accuracy (Design Input 3.4). The grid bar end pieces activity is not increased by 15% because of conservatism inherent in source term determination as indicated in Reference 2.19. The ISOSHLD-PC/QAD code does not have radionuclide source term data files for Ni-59 and Ni-63. Ni-59 and Ni-63 will be treated as Fe-55 and C-14 respectively in the QAD evaluation. Ni-63, and C-14 are beta emitters. They have average beta energies respectively of 0.0171, and 0.0495 MeV per decay [Design Input 3.14]. Ni-59 and Fe-55 are low energy gamma emitters. They have average gamma energies respectively of 0.0024, and 0.0017 MeV per decay [Design Input 3.14]. Ni-63 is treated as C-14. The equivalent amount of C-14 from Ni-63 is determined with consideration given to the relative beta energies. The equivalent C-14 activity is equal to the Ni-63 activity multiplied by the ratio of the Ni-63 average beta energy to the C-14 average beta energy i.e., (0.0171/0.0495). The equivalent amount of Fe-55 from Ni-59 is determined with consideration given to the relative gamma energies. The equivalent Fe-55 activity is equal to the Ni-59 activity multiplied by the ratio of the Ni-59 average gamma energy to the Fe-55 average gamma energy i.e., (0.0024/0.0017).

The following components occupy positions both within and outside the active fuel elevation (reactor cladding/vessel wall, reactor insulation, thermal shield, thermal shield retainers, and thermal shield seal weights). When modeling the active core region of these components all of the activity was assumed to be located within the active fuel region. Activities in component sections above or below the active fuel region were determined by multiplying the component's total activity by a source correction factor and by the ratio of the axial length of the component section outside the active fuel region to the axial length in the active fuel region. The source correction factor is determined based on the axial thermal neutron flux distribution. Source correction factors are determined in Section 6.4.

Per Table 3-1 of Design Input 3.2, the reactor vessel wall cladding activity is 245 Curies. The cladding activity was added to the vessel wall activity and modeled in the vessel wall region in the QAD model. To account for attenuation of the cladding activity by the 5.25" thick vessel wall, the cladding activity was reduced by a factor of 30 (Design Input 3.13).

There are 6 thermal shield retainers. The QAD model determined the dose rates from 1 retainer which extended from 84.4 to 95.6 degrees with side dose points located at 90 degrees (Section 6.4). QAD determined side dose rates at 2 meters were multiplied by 3 and top/bottom centerline dose rates by 6 to accommodate dose contributions from multiple retainers. Contact side dose rates were not increased.

There are 12 thermal shield seal weights. The seal weights were modeled in QAD as One plate 36" wide by 81" long and 1" thick containing half the total seal weight activity. The angular boundaries for the seal weight model were set between 70 and 110 degrees (Section 6.4). QAD determined top/bottom centerline dose rates were multiplied by 2 to accommodate dose contributions from all the seal weights.

The activities for the inlet diffuser (2.24E-2 Curies) and the inlet baffle (1.63E-1 Curies) were added to the core support plate activity (1.4E-1 Curies). These components were thus effectively modeled as part of the core support plate. Adding the activities together and modeling these components as part of the core support plate is conservative in that it increases the activity at the very bottom of the reactor vessel and thus maximizes the dose rates to areas below the reactor vessel. The activity for these 3 components was increased by 1.0 Curie to account for radioactive crud which could be present at the inside bottom of the vessel (Assumption 4.7). The total activity was rounded up to 2.0 Curies. The roundup encompasses any activation of the bottom portion of the RV wall.

The adjusted component radionuclide activities presented in Table 6-1 are used in conjunction with the radionuclide distribution fractions given in Table 3-2 to obtain the component source term activities. When applicable, a source correction factor is also applied to the component inventories to account for axial source term distribution. Radionuclide activities used in each component are provided in Attachment J to this calculation.

5.2.4 Dose Points

Dose Point locations are described and coordinates provided in Attachment J to this calculation. Contact and 2 meter dose point locations are shown in Figure J-1.

5.3 Reactor Vessel Nozzle Penetration Model

The reactor vessel includes penetrations associated with nozzles. The largest of these nozzle penetrations are 20" diameter recirculating water inlet penetrations located near the bottom of the vessel and 14" diameter steam outlet penetrations located near the top of the vessel (Design Input 3.12). Dose rates outside the transport package due to radiation streaming through these nozzle penetrations were determined using the ISOSHLD-PC computer code [Reference 2.8] with a cylindrical source with end shields geometry. The 20" diameter penetration is sufficiently below the active core region of the reactor vessel and the 14" diameter penetration sufficiently above the active core region that penetration streaming dose rates due to active core region sources would not be significant. The distances between these penetrations and the active core region sources are in the range of 8' which provides an effective thickness of the in-vessel LDCC between the penetrations and these sources of about 6" of iron equivalent. The radiation source terms that are considered in the penetration radiation streaming model are those in the vicinity of

the penetrations. These sources are the steam baffle and the sparger for the 14" diameter penetration and the inlet baffle, and the inlet diffuser for the 20" diameter penetration. The calculated penetration streaming dose rates were added to the QAD determined dose rates at dose points in the vicinities of these penetrations. The relevant dose points in the vicinities of these penetrations are those located along the side and near the top of the transport package for the 14" diameter nozzle penetrations. For the 20" diameter nozzle penetrations, the relevant dose points are located along the side near the bottom of the transport package and dose points located off the bottom of the transport package along the reactor vessel inside diameter.

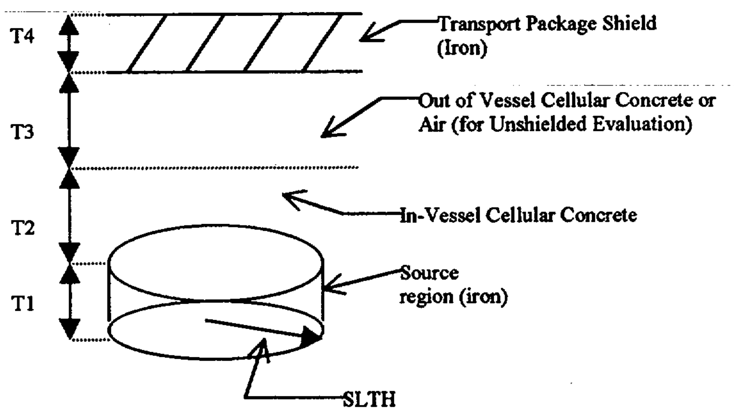
Input and Output files for the ISOSHLD models are identified in Attachment I to this calculation.

5.3.1 20" Diameter Recirculating Water Inlet Nozzle

The dose rate due to shine through this nozzle penetration is determined using the ISOSHLD-PC cylindrical source – slab shields on cylinder end geometry i.e., IGEOM = 9. The radiation source term is due to all the activity in one of the two inlet diffusers and a portion of the activity in the inlet baffle. Dose rates are determined separately for the inlet diffuser source and the inlet baffle source and are added together to obtain the final dose rate. Dose rates are determined due to nozzle penetration shine at the side of the vessel (opposite the penetration) and below the vessel i.e., underneath the penetration. For the transport package shielded case, dose rates are maximum at the side of the transport package due to the shorter distances between the source and the transport package shield. These side dose rates are applied to both the side and bottom of the transport package.

Relevant distances e.g., between the 20" diameter nozzle penetration and the side and bottom of the transport package are calculated in Section 6.5. The distances are factored into the computer model.

ISOSHLD-PC Computer Model:
IGEOM = 9



SLTH = radiation source radius = 6.875" for inlet diffuser source (Section 6.6). When addressing the inlet diffuser source, the source cylinder in the ISOSHLD model is sized such that the cross sectional area of the source is equal to the equivalent diffuser area.
= 10" for inlet baffle source (Since the inlet baffle size exceeds the diameter of the penetration, the effective radius of the source term equals the penetration radius).

T1 = source term depth = 0.375" i.e., plate thickness for inlet diffuser source
= 0.625" i.e., thickness for inlet baffle source
(0.375" and 0.625" are from Design Input 3.8)

T2 = distance between reactor vessel inside radius and nozzle tip
= 11.94" (Section 6.5)

T3 = distance between nozzle tip and transport package inner radial wall
= 10" (Section 6.5)

T4 = Transport package radial wall thickness in vicinity of penetration
= 3." (Assumption 4.5)

X = the distance between the far side of the source cylinder and the dose point. This is T1 + T2 + T3 + T4 + remaining distance to dose point.
= 130.4", 26.3", and 104" at 3 meters (unshielded), contact to transport container and 2 meters from transport container respectively for inlet diffuser source.
= 130.7", 26.6", and 104.3" at 3 meters (unshielded), contact to transport container and 2 meters from transport container respectively for inlet baffle source.

5.3.2 14" Diameter Steam Nozzle

The ISOSHLD-PC computer model geometry for the 14" diameter nozzle penetration evaluation is the same as that depicted for the 20" diameter nozzle penetration evaluation. All that changes is the numerical values for the parameters. Parameters specific to the 14" diameter nozzle follow.

SLTH	= radiation source radius = 7" i.e., nozzle penetration radius
T1	= source term depth = 1" based on steam baffle thickness (Design Input 3.5)
T2	= distance between reactor vessel inside radius and nozzle tip = 5.25" i.e., reactor vessel wall thickness (for situation where nozzle is cut down to reactor vessel surface)
T3	= distance between nozzle tip and transport container inner radial wall = 16.53" (Section 6.5)
T4	= Transport container radial wall thickness in vicinity of penetration = 3." (Assumption 4.5)
X	= the distance between the far side of the source cylinder and the dose point. This is T1 + T2 + T3 + T4 + remaining distance to dose point. = 124.36", 26.78", and 104.52" at 3 meters (unshielded), contact to transport container and 2 meters from transport container respectively.

5.3.3 Nozzles N-12 and N-14

Nozzle penetrations N-12 and N-14 are above and closest to the active core region of the reactor vessel. These 3" diameter penetrations are also closest to the grid bar end pieces (Design Input 3.12). Section 6.8 of this calculation demonstrates that there is no direct radiation shine through these penetrations from grid bar end pieces located in the immediate vicinity of the penetrations. There could be direct radiation shine through the penetrations from grid bar end pieces located on the opposite side of the reactor vessel wall from the penetration. For this situation the internal vessel LDCC provides an equivalent iron shielding thickness of about 5.5" which reduces the direct dose between the grid bar end pieces and the penetration inlet significantly (i.e., greater than an order of magnitude).

5.4 Radiation Streaming through LDCC Fill Hole Plugs in Transport Cask

As indicated in Section 3.15, holes (3" diameter (+/- 1/8")) are provided in the 3" thick transport container annular side shield in order to inject LDCC into the annular area between the reactor vessel and the transport container. There will also be 10 holes, each of 1-1/2" diameter (+/- 1/16") in the 4" thick transfer container cover. The holes are to be filled with a metal plug threaded into position. A plug length of 3" will be used for plugs to be inserted into the transfer container side and top shields. The ISOSHLD-PC computer code geometry model described for the 20" diameter nozzle penetration (i.e., Geometry 9, slab shields on cylindrical source end) is used to model radiation streaming through the plugged LDCC fill holes. The model determines the radiation streaming dose rate through the 3" diameter plugged LDCC fill hole on the side of the transfer container. The LDCC fill hole that results in the largest dose rates is used (i.e., the LDCC fill hole that is in line with the greatest radiation source inventory). A plug diameter of 3-1/8" and a plug length of 2" i.e., 1" less than the thickness of the transfer container annular shield wall is used in the computer model. The resulting dose rates bound the dose rates that could be

expected above the transport container due to the 1-1/2" diameter LDCC fill holes plugged with metal plugs having a length 1" less than the transfer cask top shield.

ISOSHLD-PC computer code model input parameters are provided in Table 5-2. The Co-60 Source term inventories in the various components are obtained from Section 6.7.3 of this calculation. Component thicknesses are obtained from information provided in Section 3 of this calculation. Partial densities for stainless steel components, for the carbon steel (CS) reactor vessel, and for RV insulation are obtained from Section 6.3 of this calculation. A density of 0.8009 g/cc concrete (ISOSHLD-PC material designator # 103) is specified for the LDCC located between the RV and the transport container (Section 5.2.2). The plugs for the LDCC fill holes in the transport container are treated as iron with a density of 7.86 g/cc.

Table 5-2: ISOSHLD-PC Input Parameters for Streaming Evaluation out RV Transport Container LDCC Fill Hole Plugs

	ISOSHLD-PC Code Input Parameters							
Source Component	Source Strength (Ci of Co-60)	SLTH (inches)	T1 (inches)	T2 (inches)	T3 (inches)	T4 (inches)	T5 (inches)	X (inches)
Contact Dose Rate:								
RV Insulation	1.77E-2	8"	3" SS,	13.5" LDCC,	2" Iron,	—	—	20.5"
RV Wall/Cladding	2.42E-1	10"	5.4" CS,	3" SS,	13.5" LDCC	2" Iron,	—	25.9"
Thermal Shield	2.74	10"	1.5" SS	5.4" CS,	3" SS,	13.5" LDCC	2" Iron,	27.4"
TS Seal Weights	2.15	10"	1" SS	5.4" CS,	3" SS,	13.5" LDCC	2" Iron,	26.9"
TS Retainer	2.56	10"	0.75" SS	5.4" CS,	3" SS,	13.5" LDCC	2" Iron,	26.65"
2 Meter Dose Rate:								
RV Insulation	1.57E-3	2"	3" SS	13.5" LDCC	2" Iron	—	—	98.24"
RV Wall/Cladding	1.23E-2	2"	5.4" CS	3" SS	13.5" LDCC	2" Iron	—	103.64"
TS Retainer	8.32E-1	2"	0.75" SS	5.4" CS	3" SS	13.5" LDCC	2" Iron	104.39"
Notes: T1 values represent the thickness of the cylindrical radiation source component.								

6.0 Calculations

6.1 Calculate the adjusted activity for each RVAI component by applying a factor of 1.15 to each activity from Table 3-1

Table 6-1. Adjusted Reactor Component Activities

Component	Activity	Adjusted Activity
Steam Baffle	1.08 E+1 Curies	1.24 E+1 Curies
Sparger	8.60 E -1 Curies	9.89 E -1 Curies
Top Guide Plate	1.14 E+3 Curies	1.31 E+3 Curies
Seal Housing	1.30 E+2 Curies	1.50 E+2 Curies
Thermal Shield (TS)	7.40 E+3 Curies	8.51 E+3 Curies
TS Retainer	1.14 E+2 Curies	1.31 E+2 Curies
Seal Weights	1.10 E+3 Curies	1.27 E+3 Curies
RV Insulation	5.81 E+1 Curies	6.68 E+1 Curies
RV Wall	1.11 E+3 Curies	1.28 E+3 Curies
RV Wall Cladding	2.45 E+2 Curies	2.82 E+2 Curies
Core Support Plate	1.40 E -1 Curies	1.61 E -1 Curies
Inlet Diffuser	2.24 E -2 Curies	2.58 E -2 Curies
Inlet Baffle	1.63 E -1 Curies	1.87 E -1 Curies
Grid Bar End Pieces	1.80 E+3 Curies	1.80 E+3 Curies*
Total	1.31 E+4 Curies	1.48 E+4 Curies

* The grid bar end pieces activity was not increased by 15% because of conservatism inherent in source term determination as indicated in Section 5.2.3.

The potential 15% under-response of the radiation monitoring instrumentation is compensated by the factor of 1.15. The potential 15% under-response is discussed in Design Input 3.4.

6.2 The conversion factor from lbs/ft^3 to g/cm^3 is obtained as follows (Design Input 3.1):

$$1 \ \text{lbs} \ / \ 1 \ \text{ft}^3 = 453.6 \ \text{g} \ / ([12 \ \text{in/ft} \times 2.54 \ \text{cm/in}]^3) = 1.60 \ E\text{-}2 \ \text{g/cm}^3 \ ;$$

6.3 Determination of Partial Densities for Carbon Steel, for 304 Stainless Steel, and for Grid Bar End Piece Source Region

Carbon and 304 Stainless Steel:

Table 6.2. Carbon Steel & 304 Stainless Steel Partial Densities (g/cc)

Material Composition Components	Weight Percents (1) 304SS	Carbon Steel	Shield Material Partial Densities 304 SS (2)	Carbon Steel (3)	Vessel Insulation (4)	Core Support Plate (5)	Carbon Steel on RV Bottom (6)
Nitrogen	0.045	0.008	3.537E-03	6.288E-04	1.442E-04	1.769E-03	5.030E-04
Chromium	18.4	0	1.446E+00	0.000E+00	5.895E-02	7.231E-01	0.000E+00
Manganese	1.53	1.35	1.203E-01	1.061E-01	4.902E-03	6.013E-02	8.489E-02
Iron	70.6	97.57	5.549E+00	7.669E+00	2.262E-01	2.775E+00	6.135E+00
Nickel	10	0.61	7.860E-01	4.795E-02	3.204E-02	3.930E-01	3.836E-02
Molybdenum	0.26	0.58	2.044E-02	4.559E-02	8.330E-04	1.022E-02	3.647E-02
Niobium	0.009	0.002	7.074E-04	1.572E-04	2.883E-05	3.537E-04	1.258E-04
Cobalt	0.141	0.012	1.108E-02	9.432E-04	4.517E-04	5.541E-03	7.546E-04

Notes to Table:
(1) Elemental weight percents for 304 SS and carbon steel are from Assumption 4.3.
(2) Partial elemental densities for 304 stainless steel are obtained by multiplying the density of 7.86 g/cc by the 304 SS weight percent for each element (column 2 of the table divided by 100). The density of 7.86 g/cc is from Assumption 4.3.
(3) Partial elemental densities for carbon steel are obtained by multiplying the density of 7.86 g/cc by the carbon steel weight percent for each element (column 3 of the table divided by 100). The density of 7.86 g/cc is from Assumption 4.3.
(4) Partial elemental densities for vessel insulation are obtained by multiplying the insulation density by the 304 SS weight percent for each element. The insulation density is (5180 lbs/259 ft^3)* 453.6 g/lb*(1/(30.48cm/foot)^3) = 0.32037 g/cc. The insulation weight and volume are from Design Input 3.11.
(5) Partial elemental densities for core support plate are obtained by multiplying the core support plate density by the 304 SS weight percent for each element. The core support plate density is assumed to be 0.5 * 7.86 g/cc = 3.93 g/cc (Assumption 4.3).
(6) Partial elemental densities for the reactor vessel bottom are obtained by multiplying the reactor vessel bottom density by the carbon steel weight percent for each element. The reactor vessel bottom density is assumed to be 0.8 * 7.86 g/cc = 6.288 g/cc (Assumption 4.3).

Grid Bar End Pieces:
The QAD zone used to model the grid bar end pieces is a 12" high semi-annular geometry whose radial dimensions are defined by the top guide plate inner and outer radial boundaries i.e., it extends from 44" to 50" (Assumption 4.4). The grid bar end pieces zone volume is thus:

$$0.5 * [12 * \pi * [(50")^2 - (44")^2] * (2.54 cc/in)^3] = 1.74E+5 \text{ cc}$$

Where The factor of 0.5 accounts for a semi-annular region from 0 to 180 degrees,
44" and 50" are the zone inner and outer radii,
12" is the zone height, and
2.54 is the inch to centimeter conversion factor (Design Input 3.1)

The partial density of the iron in this zone is:

[(171 lbs * 453.6 grams/lb)/1.74E+5 cc] = 4.457E-1 grams/cc

Where 171 lbs is the mass of the grid bar end pieces (Design Input 3.11), and
453.6 = grams per pound conversion factor (Design Input 3.1)

Void spaces between grid bar end pieces are filled with in-vessel LDCC with a density of 30 lb/ft^3 (0.4806 grams/cc).

The volume occupied by the grid bar end pieces (using a density of 7.86 grams/cc for the end pieces) is:

[(171 lbs * 453.6 grams/lb)/7.86 grams/cc] = 9.868E+3 cc

The volume occupied by the LDCC within the zone is:

(1.74E+5 cc – 9.868E+3 cc) = 1.64E+5 cc

The partial density of LDCC is thus:

0.4806 grams/cc * (1.64E+5 cc/1.74E+5cc) = 4.529E-1 grams/cc

The total density in this zone is the sum of the iron and LDCC partial densities:

(0.4457 grams/cc + 0.4529 grams/cc) = 0.8986 grams/cc

6.4 QAD Model Source Correction Factors:

The source correction factors are used to obtain the activity concentrations for components that are partially located in the active fuel region and partially out of the active fuel region. The source correction factors are obtained using the axial thermal neutron flux values presented in Table 3-7 in conjunction with an interpolation technique. The source correction factors are multiplied by the total activity of the component of concern and by the ratio of the length of the segment under consideration that lies outside the active fuel region to the length within the active fuel region to obtain the activity in the section of the component which is outside the active fuel region. Source correction factor calculations follow:

- From Bottom of Active Fuel Region to 4" Below Active Fuel Region

  A source correction factor of 0.5 is applied (Design Input, Table 3-7).

- From 4" Below Active Fuel Region to 15" Below Active Fuel Region

  Thermal neutron flux at 4" below active fuel region: 2E+12 (Design Input, Table 3-7)
  Thermal neutron flux at 15" below active fuel region: 8E+11 (Design Input, Table 3-7)

  Semi-log interpolation is used to obtain the flux value at a distance half-way between 4" and 15" below the active fuel i.e., (4 + 15)/2 = 9.5" below active fuel.

$[(\ln(x) - \ln(2E+12))/(\ln(8E+11) - \ln(2E+12))] = 0.5$

Solving for 'x' gives 'x' = 1.26E+12
The source correction factor is (1.26E+12)/(1.5E+13) = **0.084**

- From 15" Below Active Fuel Region (i.e., Bottom of TS Retainer) to 20" Below Retainer

  Thermal neutron flux at bottom of TS retainer: 8E+11 (Design Input, Table 3-7)

  Thermal neutron flux at 20" below TS retainer: 2E+9 (Design Input, Table 3-7)

  Semi-log interpolation gives a flux value of 4E+10 at the half-way distance and a source correction factor of **2.7E-3**

- For Distances Greater Than 20" Below TS Retainer

  The thermal neutron flux in this location is orders of magnitude less than the flux at the active core region. The radiation source term activation concentrations are also thus orders of magnitude less than they are in the active fuel region. The source correction factor at this distance below the active fuel region was determined using linear interpolation. Linear interpolation results in a larger source correction factor and is conservative.

  Thermal neutron flux at 20" below TS retainer: 2E+9 (Design Input, Table 3-7)
  Thermal neutron flux at core support plate elevation: 1E+5 (Design Input, Table 3-7)

  Linear interpolation gives a flux value of [(2E+9) + (5E+5)]/2 = 1E+9. The source correction factor is thus (1E+9)/(1.5E+13) = **6.7E-5**

- From Top of Active Fuel Region to Top of Thermal Shield

  Thermal neutron flux at active fuel region: 1.5E+13 (Design Input, Table 3-7)
  Thermal neutron flux at top of thermal shield: 5E+12 (Design Input, Table 3-7)

  Semi-log interpolation gives a flux value of 8.66E+12 at the half-way distance and a source correction factor of **0.577**.

- From Top of Thermal Shield to 20" Above Thermal Shield

  Thermal neutron flux at top of thermal shield: 5E+12 (Design Input, Table 3-7)
  Thermal neutron flux at 20" above thermal shield: 2E+11 (Design Input, Table 3-7)

  Semi-log interpolation gives a flux value of 1E+12 at the half-way distance and a source correction factor of **0.067**

- From 20" Above Top of Thermal Shield to Steam Baffle Elevation

  Thermal neutron flux at 20" above top of TS: 2E+11 (Design Input, Table 3-7)
  Thermal neutron flux at steam baffle elevation: 2E+9 (Design Input, Table 3-7)

Semi-log interpolation gives a flux value of 2E+10 at the half-way distance and a source correction factor of **1.33E-3**

- For Distances Above the Steam Baffle

Linear interpolation is used.
Thermal neutron flux at steam baffle elevation: 2E+9 (Design Input, Table 3-7)
Thermal neutron flux at RV flange elevation: 5E+7 (Design Input, Table 3-7)

Linear interpolation gives a flux value of [(2E+9) + (5E+7)]/2 = 1.025E+9. The source correction factor is thus (1E+9)/(1.5E+13) = **6.83E-5**

- For Thermal Shield Seal Weights

There are 12 seal weights which are located between the thermal shield and the reactor vessel wall (Design Input 3.7). The activity in each seal weight is 1/12th the total activity in all seal weights. The width of each seal weight plate is 18" (Design Input 3.7). Two seal weights side by side are thus about 2 x 18" = 36" wide. The angle subtended by 2 seal weights positioned side by side is determined as the width of the seal weights divided by the circumference at the seal weight location times 360 degrees i.e.,

$$360^\circ \times [(2 \times 18")/(3.14159 \times (103"))] = 40.0^\circ$$

For a seal weight source consisting of 2 seal weights located side by side and centered on 90°, the source term angle would extend from 70° to 110°.

- For Thermal Shield Retainer

There are 6 thermal seal retainers which are located between the thermal shield and the reactor vessel wall (Design Input 3.7). The activity in each retainer is thus 1/6th the total activity in all retainers. The width of each retainer is 10" (Design Input 3.7). The angle subtended by a retainer is determined as the width of the retainer divided by the circumference at the retainer location times 360 degrees i.e.,

$$360^\circ \times [(10")/(3.14159 \times (103"))] = 11.1250^\circ$$

For a retainer centered on 90°, the source term angle would extend from 84.4° to 95.6°

6.5 Nozzle Penetration Distances (Drawing F-230-791-2, Reference 2.13)

<u>20" Diameter Inlet Nozzle</u>:

Distance from reactor vessel tangent line near vessel bottom to reactor vessel inside surface at vessel bottom: 53 & 7/32"
Distance from reactor vessel tangent line to end of 20" nozzle: 65 & 27/32"
The angle for the nozzle centerline is 20 degrees downward from the reactor vessel tangent line near the bottom of the vessel.
Distance from reactor vessel centerline to transport package inside side wall:
((13' x 12"/ft)/2) – 3" =75"

Where 13' and 3" are transport package radial diameter and wall thickness respectively per Assumption 4.12

Distance from reactor vessel tangent line to inside surface of transport package bottom: (53 & 7/32" + 5.25" + 4") =62 & 15/32"

Where 5.25" = reactor vessel wall thickness (Design Input 3.7) and 4" = minimum distance between bottom of reactor vessel and inside bottom surface of transport package (Assumption 4.5)

The inside circumference for the bottom hemisphere of the reactor vessel is:

(0.5) x π x (2 x 53 & 7/32") = **167.2"**

The arc length from the tangent line to the far end of the nozzle opening is thus:

(20°/180° x 167.2") + 10" = **28.6"** (Note: 10" is nozzle radius)

The angle from the tangent line to the far end of the nozzle opening is thus:

(28.6"/167.2") x 180° = **30.8°**

Distance between end of nozzle and reactor vessel centerline is:

X = (65 & 27/32") x cos(9.3°) = 65" (Note: 9.3° angle is shortest angle between tangent line and closest point on nozzle i.e., (18.6" – 10")/167.2" x 180° = 9.3°)

The distance between the nozzle and the inside radial surface of the transport package is thus:

(75" – 65") = **10"**

Distance between reactor vessel bottom tangent line and bottom of nozzle is:

Y = (65 & 27/32") x sin(30.8°) = **33.7"**

The distance between the nozzle and the inside bottom surface of the transport package is thus:

(62 & 15/32") – 33.7" = **28.8"**

Distance between reactor vessel inside radius and the tip of the nozzle is about:

(65" – 53.0625") = **11.94"**

<u>14" Diameter Steam Nozzle</u>:

Distance between inside radial surface of transport package and external radial surface of reactor vessel is:

((13' x 12"/ft)/2 – 3" – 58.47") = **16.53"**

Where 13' = transport package outer diameter (Assumption 4.5)
3" = transport package radial wall thickness (Assumption 4.5)
58.47" = reactor vessel outer radius (Design Input 3.7)

6.6 Radiation Source Terms for Nozzle Streaming Evaluations

20" Diameter Inlet Nozzle:

There are 2 source components i.e., activity in the inlet diffuser and activity in the inlet baffle.

The inlet diffuser source is 27" by 5.5" by 3/8" thick (Design Input 3.8). A 3/8" cylinder with an equivalent cross sectional area results in a cylinder radius of:

$[(27" \times 5.5")/\pi]^{1/2}$ = **6.875"**

Since there are 2 inlet diffusers, one in line with each of 2 nozzle penetrations, half the inlet diffuser total activity is used for the nozzle penetration.

The inlet baffle is located in the vicinity of the reactor vessel internal circumference (Design Input 3.8). The fraction of the inlet baffle activity available for shine through the 20" diameter penetration is taken as the ratio of the penetration diameter (i.e., 20") to the reactor vessel internal circumference i.e.,

$20"/(\pi \times (2 \times 53.22"))$ = **5.98E-2**

Where 53.22" = reactor vessel wall inner radius (Design Input 3.7)

The inlet baffle and inlet diffuser nuclide specific radiation source term activities available for shine through the penetration are presented in Attachment J to this calculation.

14" Diameter Steam Nozzle:

Radiation sources in the steam baffle and the sparger contribute to the dose rate due to shine out the 14""diameter penetration.

The sparger is a 2" diameter pipe ring located around the inner circumference of the reactor vessel (Design Input 3.5). The fraction of the sparger activity available for shine through the 14" diameter penetration is taken as the ratio of the penetration diameter (i.e., 14") to the reactor vessel inner diameter i.e.,

$14"/(2 \times 53.22")$ = **0.132**

Where 53.22" = reactor vessel wall inner radius (Design Input 3.7)

The fraction of the steam baffle source considered to contribute to dose rates due to shine out the penetration is taken as the ratio of the penetration surface area to the steam baffle surface area. The steam baffle surface area is assumed to cover 80% of the reactor vessel internal cross sectional area (Assumption 4.15). The radiation source fraction is thus:

$[(\pi \times (7)^2)/(\pi \times 0.8 \times (53)^2)]$ = **0.022**

The sparger and steam baffle nuclide specific radiation source term activities available for shine through the penetration are presented in Attachment J to this calculation.

6.7 Calculations Related to LDCC Fill Holes

6.7.1 Spatial Relationship between LDCC Fill Holes and RV Active Fuel Region

Section 3.15 indicates that the 4 LDCC fill holes in the side wall of the transfer container are located 3' (i.e., 36") from the transfer container's top surface, 6' (i.e., 72") from the transfer container's top surface, 3'-6.5" (42.5") from the transfer container's bottom surface, and 8'-2.6875" (98.6875") from the container's bottom surface.

From Figure H-1 of Attachment H to this calculation it can be seen that the top surface of the transport container (153.25" above reactor core midplane) is 118.25" (i.e., 153.25" – 35") above the top of the active fuel region and 102.25" (i.e., 153.25" – 51") above the transport container 7" to 3" side shield transition. The distances between the RVAI major radiation source term components and the top surface of the transport container are equal to or greater than the distances between RVAI major radiation source term components and the 7" to 3" side shield transition. The 2 LDCC fill holes near the top of the transport container are at least 46.25" (i.e., 118.25" – 72") above the active fuel region and 30.25" above the transport container wall 7" to 3" transition.

From Figure J-1 of Attachment J to this calculation, it can be seen that the bottom surface of the transport container is 8'-5.375" below the transport container 7" to 3" side shield transition. The bottom of the 7" to 3" side shield transition is aligned with the bottom of the thermal shield which is 10" below the bottom of the active fuel region. Thus the bottom surface of the transport container is 9'-3.375" (111.375") below the bottom of the active fuel region. The 2 LDCC fill holes near the bottom of the transport container are 68.875" (i.e., 111.375" – 42.5") below the bottom of the active fuel region and 12.6875" (i.e., 111.375" – 98.6875") below the bottom of the active fuel region.

The RVAI radiation source components that are significant contributors to the dose rates around the transport container and thus significant contributors to the dose rates due to radiation shine through the LDCC fill hole plugs are the grid bar end pieces and source components in the RV active core region. From the above evaluation, it is determined that the closest LDCC fill hole to the major radiation source components is the LDCC fill hole centered 12.6875" below the active fuel region.

6.7.2 LDCC Fill Hole Radiation Source Component Dimensions

Figure H-1 of Attachment H to this calculation indicates:

- radial distance to the external surface of the transport container shield wall is 78";
- radial distance to the internal surface of the transport container shield wall is 75";
- radial distance to the external surface of the RV insulation is 61.46875";
- radial distance to the external surface of the RV wall (RV insulation internal radius) is 58.46875";
- radial distance to the internal surface of the RV wall cladding is 53.0625";
- transfer container shield wall thickness below RV active fuel region is 3" (i.e., 78"-3")
- RV insulation thickness is 3" (i.e., 61.46875" – 58.46875")

- RV wall/cladding thickness is 5.25" + 5/32" = 5.4" (Section 3.7)

Source Term Component Radial Dimensions for Contact Dose Point

- The contact dose point is set at 1" from the transport container surface:
- The radial dimension of the LDCC fill hole is: (3.125"/2) = 1.5625" (Section 3.15)
- The radial dimension (R) for RV insulation source term segment which could contribute to radiation streaming out the plugged LDCC fill hole at a contact dose point can be determined using similar triangles:

  R/(1" + 3" + 13.5" + 3") = 1.5625"/4"

  R = 8.0"

  Where

  3" is the thickness of the RV transfer cask shield and the thickness of the RV insulation (Section 3.7),

  4" is the sum of the transfer cask wall thickness and 1" contact dose point distance.

  13.5" is the LDCC thickness between RV insulation and the transport container (75"-61.46875")

Similarly, the radial dimensions for the RV wall and source term components within the RV confines are determined to be 10".

Source Term Component Radial Dimensions for 2 Meter Dose Point

- The dose point is two meters from the transport container surface:

  (2*100 cm)/(2.54 cm per inch) = 78.74"
- The radial dimension of the LDCC fill hole is: (3.125"/2) = 1.5625" (Section 3.15)
- The radial dimension (R) for a source term component located at the inside surface of the RV wall and which could contribute to radiation streaming out the plugged LDCC fill hole at a 2 meter dose point can be determined using similar triangles:

  R/(78.74" + 3" + 13.5" + 3" + 5.25") = 1.5625"/81.74"

  R = 1.98" (approximately 2")

  3" is the thickness of the RV transport container shield and the thickness of the RV insulation (Section 3.7)

  5.25" is the thickness of the RV wall (Section 3.7)

  81.74" is the sum of the transport container wall thickness and the 2 meter dose point distance of 78.74"

6.7.3 Source Term Identification and Inventories for LDCC Fill Hole Source Components

The major RVAI radiation source components are along the RV active fuel region or slightly above or below the active fuel region. As indicated in Section 6.7.1, the closest LDCC fill hole to the active fuel region is the LDCC fill hole centered at 12.6875" below the active fuel region. For dose rate determination, only the Co-60 radiation source term will be considered in this section because Co-60 is the only significant source term contributor to the dose rate (Assumption 4.1).

Section 6.7.2 indicates that the source term segments which contribute to the radiation streaming dose rate out the LDCC fill hole plug are cylindrical in geometry. The Co-60 activity in the cylindrical source term component is determined by multiplying the total activity in the source component by the ratio of the area in the cylindrical source segment to the total area of the

source component. The Co-60 concentration for RV insulation, the thermal shield, and the RV wall/cladding varies as a function of elevation below the active fuel region and changes within the elevation span of the cylindrical source segment that shines through the LDCC fill hole plug. The total Co-60 source in the cylindrical source segment from each source component (i.e., the RV insulation and the RV wall/cladding) is determined and homogenized over the cylindrical source segment surface area. The varying Co-60 concentrations in the cylindrical source segment due to changes in Co-60 concentrations in the RV insulation, thermal shield, and RV wall/cladding is accounted for by defining the surface area of the cylindrical source segment covered by the various source concentrations. The area is determined using the following equation:

$$A_1=\{(\frac{2\theta}{360})\Pi R^2\}-h_1R(\sin\theta) \qquad \text{Equation 6.1}$$

Where: The variables are as described in Figure 6.1, specifically,

θ = arccos of h_1/R
R = radius of cylindrical source segment that shines out LDCC fill hole
360 = 360° i.e., angle for a full circle
$h_1 = R - h_2$ where h_2 is the length over which the component source region covers the cylindrical source segment surface area
$h_1R(\sin\theta)$ = L (shown in Figure 6.1)
A1 = the area above the chord 'A – B'

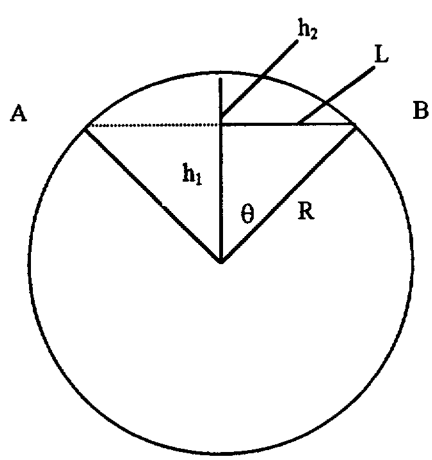


Figure 6.1

Contact Dose Point

As indicated in Section 6.7.2, the radial dimension for the cylindrical source segments, which contribute to the contact streaming dose rate, is 8" for RV insulation and 10" for all other source components. The source region surface area is thus; (3.14159* (8")^2) = 201.1 square inches for the 8" diameter source segment and (3.14159* (10")^2) = 314.2 square inches for the 10" diameter source segment. The centerline for the closest LDCC fill hole to the active fuel region is 12.6875" below the bottom of the active fuel region (Section 6.7.1). Thus the components with significant

radiation source terms which would contribute to the streaming dose rate through this LDCC fill hole plug are those that are between 2.6875" and 22.6875" below the active fuel region (4.6875" and 20.6875" for RV insulation). These components consist of RV insulation, RV wall/cladding, RV thermal shield, RV seal weights, and the bottom portion of the thermal shield retainer (determined from information presented in Section 3 and Attachment J).

RV Insulation

Below the active fuel region, the RV insulation source region boundaries occur at 4" below the active fuel region, at 15" below the active fuel region, and at 35" below the active fuel region (Section 3.7 and Attachment J). Since the LDCC fill hole cylindrical source region extends from 4.6875" to 20.6875" below the active fuel region, only the RV insulation source segments from 4" to 35" below the active fuel region contribute to the Co-60 inventories.

From 4" to 15" below the active fuel region there are 0.504 Curies of Co-60 in RV insulation (Attachment J). The 0.504 Curies are spread over an area of:

$$\pi(58.46875")*(2)*(15"-4") = 4.04E+3 \text{ square inches}$$

From 15" to 35" below the active fuel region there are 2.95E-2 Curies of Co-60 in RV insulation (Attachment J). The 2.95E-2 Curies are spread over an area of:

$$\pi(58.46875")*(2)*(35"-15") = 7.35E+3 \text{ square inches}$$

The parameters for the areas of the LDCC fill hole 8" radial cylindrical source covered by the RV insulation source segments are determined using Equation 6.1 and are provided in the following table.

Table 6-2-1. LDCC Fill Hole Parameters for 8" Source Covered by RV Insulation Source Segments

Parameter	RV Insulation 4" to 15" Below Active Fuel	RV Insulation 15" to 35" Below Active Fuel
R (inches)	8	8
h_1 (inches)	(8+4.6875-15) = -2.3125 which is 2.3125" below the 8" cylindrical source centerline.	2.3125" below the 8" cylindrical source centerline
h_2 (inches)	(R- h_1) = 5.6875	5.6875
θ (degrees)	73.2	73.2
Area (A1) inches^2	Since the source extends to 2.3125" below the cylindrical source centerline, the area of the source is the area of the cylinder – A1. = (201.1 – 64.05) = 137.05	64.05

The LDCC fill hole source Co-60 activity due to the RV insulation segment from 4" to 15" below the active fuel is:

$$0.504 \text{ Curies} * (137.05/4.04E3) = 1.71E-2 \text{ Curies}$$

The LDCC fill hole source Co-60 activity due to the RV insulation segment from 15" to 35" below the active fuel is:

2.95E-2 Curies * (64.05/7.35E3) = 2.57E-4 Curies

The total Co-60 Curies in the LDCC fill hole cylindrical source attributed to the RV insulation is:

1.71E-2 + 2.57E-4 = **1.74E-2** Curies which is homogenized over the LDCC fill hole streaming cylindrical source for the ISOSHLD-PC computer model.

<u>RV Wall/Cladding</u>

Below the active fuel region, the RV wall/cladding source region boundaries occur at 4" below the active fuel region, at 15" below the active fuel region, and at 35" below the active fuel region (Section 3.7 and Attachment J). In the vicinity of the LDCC fill hole cylindrical streaming source, the RV wall/cladding source regions that contribute to radiation streaming through the LDCC fill hole plug are those from the bottom of the active fuel region to 4" below, from 4" to 15" below the active fuel region, and from 15" to 35" below the active fuel region (Section 3.7 and Attachment J). The Co-60 inventories in the RV wall/cladding segments are 7.74 from the active fuel region to 4" below, 3.58 Curies from 4" to 15" below, and 2.09E-1 Curies from 15" to 35" below (Section 6.4 and Attachment J). The 7.74 Curies are spread over an area of:

π(53.0625")*(2)*(4") = 1.33E+3 square inches

where 53.0625" = RV wall inner radius
53.0625"*(2) is the internal diameter of the RV wall/cladding (Table H-2 of Attachment H) and 4" is the height of the RV wall/cladding section.

From the 4" to 15" below the active fuel region there are 3.58 Curies of Co-60 in RV wall/cladding (Attachment J). The 3.58 Curies are spread over an area of:

π(53.0625")*(2)*(15"-4") = 3.67E+3 square inches

From the 15" to 35" below the active fuel region there are 2.09E-1 Curies of Co-60 in RV wall/cladding (Attachment J). The 2.09E-1 Curies are spread over an area of:

π(53.0625")*(2)*(35"-15") = 6.67E+3 square inches

The parameters for the areas of the LDCC fill hole 10" radial cylindrical source covered by the RV wall/cladding source segments are determined using Equation 6.1 and are provided in the following table.

Table 6-2-2. LDCC Fill Hole Parameters for 10" Source Covered by RV Wall/Cladding Source Segments

Parameter	RV Wall/Clad Segment to 4" Below Active Fuel	RV Wall/Clad 4" to 15" Below Active Fuel	RV Wall/Clad 15" to 35" Below Active Fuel
R (inches)	10	10	10
h_1 (inches)	(10+2.6875-4) = 8.6875	(10+2.6875-15) = 2.3125" below cylindrical source centerline	2.3125" below cylindrical source centerline
h_2 (inches)	(R- h_1) = 1.3125	(R- h_1) = 7.6875	(R- h_1) = 7.6875
θ (degrees)	29.7°	76.6°	76.6°
Area (A1) inches^2	8.79	A1 = 111.2. Area of source band in middle of source cylinder = (314.2 – 111.2 – 8.79) = 194.2	A1 = 111.2. = Area of source band at bottom of source cylinder

The LDCC fill hole source Co-60 activity due to the RV wall/cladding segment from the bottom of active fuel to 4" below is:

7.74 Curies * (8.79/1.33E3) = 5.12E-2 Curies

The LDCC fill hole source Co-60 activity due to the RV wall/cladding segment from 4" to 15" below the active fuel region is:

3.58 Curies * (194.2/3.67E3) = 1.90E-1 Curies

The LDCC fill hole source Co-60 activity due to the RV wall/cladding segment from 15" to 35" below the active fuel region is:

2.09E-1 Curies * (111.2/6.67E3) = 3.48E-3 Curies

The total Co-60 Curies in the LDCC fill hole cylindrical source attributed to the RV wall/cladding is:

5.12E-2 + 1.90E-1 + 3.48E-3 = **2.45E-1** Curies which is homogenized over the LDCC fill hole streaming cylindrical source for the ISOSHLD-PC computer model

Thermal Shield

The thermal shield extends to 10" below the active fuel region (Section 3.7). The Co-60 concentration in the thermal shield changes at 4" below the active fuel region. From the bottom of the active fuel region to 4" below there are 139 Curies of Co-60 in the thermal shield (Section 6.4 and Attachment J). The 139 Curies are spread over an area of:

π(50")*(2)*(4") = 1.26E+3 square inches

where 50"*(2) is the internal diameter of the thermal shield (Section 3.7) and 4" is the height of the thermal shield section.

From the 4" to 10" below the active fuel region there are 35 Curies of Co-60 in the thermal shield (Section 6.4 and Attachment J). The 35 Curies are spread over an area of:

π(50")*(2)*(6") = 1.88E+3 square inches

The parameters for the areas of the LDCC fill hole 10" radial cylindrical source covered by the thermal shield source segments are determined using Equation 6.1 and are provided in the following table.

Table 6-2-3. LDCC Fill Hole Parameters for 10" Source Covered by Thermal Shield Source Segments

Parameter	TS Segment to 4" Below Active Fuel	TS Segment 4" to 10" Below Active Fuel
R (inches)	10	10
h_1 (inches)	(10+2.6875-4) = 8.6875	(10+2.6875-10) = 2.6875
h_2 (inches)	(R- h_1) = 1.3125	(R- h_1) = 7.3125
θ (degrees)	29.7°	74.4°
Area (A1) inches^2	8.79	(104-8.79) = 95.2

The LDCC fill hole source Co-60 activity due to the thermal shield segment from the bottom of active fuel to 4" below is:

139 Curies * (8.79/1.26E3) = 0.97 Curies

The LDCC fill hole source Co-60 activity due to the thermal shield segment from 4" to 10" below the active fuel region is:

35 Curies * (95.2/1.88E3) = 1.77 Curies

The total Co-60 Curies in the LDCC fill hole cylindrical source attributed to the thermal shield is:

0.97 + 1.77 = **2.74** Curies which is homogenized over the LDCC fill hole streaming cylindrical source for the ISOSHLD-PC computer model.

TS Retainer

There are 6 TS retainers. The TS retainer extends to 5" below the bottom of the TS (Section 3.7). It thus extends to 15" below the active fuel region. The TS retainer is about 5" wide at the bottom half (Section 3.7). The Co-60 activity in the portion of 1 TS retainer that lies below the active fuel region is 3.12 Curies (Attachment J to this calculation). The portion of the TS retainer Co-60 source in the cylindrical source segment region is:

[(15" – 2.6875")/15]*(3.12 Curies) = **2.56 Curies.**

TS Seal Weight

As indicated in Section 3.7, the TS seal weights extend to the bottom of the thermal shield which is 10" below the active fuel region and are 18" thick. The TS seal weight source region that shines through the LDCC fill penetration is modeled as a 10" radial cylinder centered at 12.6875" below the active fuel region. The radial extent of the source cylinder is thus 12.8675" – 10" = 2.6875" below the active fuel region. The seal weight extends to 10" below the active fuel region and thus overlaps the cylindrical source region by 10" – 2.6875" = 7.3125" which is the value of h_2 in Figure 6.1. The value of h_1 is R-h_2 which is 10" – 7.3125" = 2.6875". The angle, θ = arccos(h_1/R) = 74.4°. The area, A1, is thus 104 square inches. The Co-60 activity in the portion of 6 seal weights located below the active fuel region is 22.3 curies (Attachment J) and the surface area is 18" wide x 10" high = 180". The amount of seal weight Co-60 activity in the region of the 10" radial cylindrical source is thus:

(22.3 Ci/6 seal weights)*(104 square in/180 square in) = **2.15 Curies.**

2 Meter Dose Point

As indicated in Section 6.7.2, the radial dimension for the cylindrical source segments which contribute to the 2 meter streaming dose rate is 2". The source region surface area is thus;

$(3.14159 * (2")^2) = 12.6$ square inches.

The centerline for the closest LDCC fill hole to the active fuel region is 12.6875" below the bottom of the active fuel region (Section 6.7.1). Thus the components with significant radiation source terms which would contribute to the streaming dose rate through this LDCC fill hole plug are those that are between 10.6875" and 14.6875" below the active fuel region. These components consist of RV insulation, RV wall/cladding, and the bottom portion of the thermal shield retainer.

The Co-60 source term activities in the 2" radial cylindrical source segment which contribute to the streaming dose rate through the LDCC fill hole plug are determined by multiplying the total source term activity in the component by the ratio of the cross sectional area of the 2" radial cylindrical source segment (i.e., 12.6 square inches) to the area of the source component.

<u>RV Insulation</u>

Co-60 inventory = 5.04E-1 Curies * [12.6/(3.14159*2(61.46875" - 3")*(15" – 4"))]
= **1.57E-3 Curies**

Where: 5.04E-1 Curies is the Co-60 RV insulation inventory from 4" to 15" below the active fuel region (Attachment J to this calculation),
(61.46875" – 3") = insulation inner radius,
(15" – 4") elevation range below active fuel region for RV insulation section.

<u>RV Wall/Cladding</u>

Co-60 inventory = 3.58 Curies * [12.6/(3.14159*2(53.0625")*(15" – 4"))]
= **1.23E-2 Curies**

Where: 3.58 Curies is the Co-60 RV wall and cladding inventory from 4" to 15" below the active fuel region (Attachment J to this calculation),
53.0625" = RV wall inner radius

<u>TS Retainer</u>

There are 6 TS retainers. The TS retainer extends to 5" below the bottom of the TS (Section 3.7). It thus extends to 15" below the active fuel region. The TS retainer is about 5" wide at the bottom half (Section 3.7). The Co-60 activity in the portion of 1 TS retainer that lies below the active fuel region is 3.12 Curies (Attachment J to this calculation). Since the width of the TS retainer is similar to the diameter of the 2" radial cylindrical source segment (i.e., 5" versus 4") and since the TS retainer extends from the top to the bottom of the cylindrical source segment, the TS retainer activity credited for streaming through the LDCC fill hole plug is conservatively taken to be proportional to the ratio of cylindrical source term segment diameter (i.e., 4") to the TS retainer segment length below the active fuel region (i.e., 15").

Co-60 inventory = 3.12 Curies * (4"/15") = **0.832 Curies**

6.8 ISOSHLD-PC and QAD Dose Rate Results

6.8.1 Dose Rates Due to Radiation Streaming through Reactor Vessel Nozzle Penetrations

ISOSHLD-PC calculated dose rates due to radiation streaming out the 14" diameter reactor vessel nozzle penetration and out the 20" diameter nozzle penetration are presented in the table that follows. The unshielded 3 meter dose rate is directly in front of the penetration and 3 meters from the reactor vessel. This dose rate was determined without taking credit for the transport container or LDCC located outside the reactor vessel however, it does take credit for in-vessel LDCC. The shielded contact and 2 meter dose rates are at contact to and 2 meters from the outer surface of the transport package. These dose rates are for dose points along the side of the transport package and directly in front of the penetration. The 20" diameter nozzle penetration is angled down 20 degrees and thus the penetration shine could contribute to dose rates on the bottom of the transport package. Application of the calculated nozzle penetration radiation streaming dose rates to other reactor vessel penetrations (of smaller diameter) and application of contact and 2 meter dose rates (calculated at the side of the transport package for the 20" diameter nozzle penetration) to the bottom edge of the transport package are conservative i.e., would overestimate the dose rates.

The radiation source term contributing to the dose rate due to shine out the reactor vessel nozzle penetrations is limited to gamma radiation (essentially Co-60). The dose rate due to scattered gammas out the penetrations from source components inside the reactor vessel but not in the penetrations' streaming paths would not be significant compared to the direct dose rates because the LDCC and transport cask shielding will significantly attenuate the low energy scattered gammas.

ISOSHLD-PC file names are identified in Attachment I to this calculation. ISOSHLD-PC input and output files are included on the CD which accompanies this calculation (Attachment M).

Table 6-3. Dose Rates due to Radiation Streaming through Reactor Vessel Penetration Nozzles

	Dose Rates (mrem/hr)		
	Unshielded	Shielded Via Transport Cask	
Penetration/Source	3 Meters	Contact	2 Meters
20" Diameter Penetration			
Inlet Diffuser Source	8.072E-2	6.435E-1	4.437E-2
Inlet Baffle Source	6.363E-2	4.490E-1	3.465E-2
Total (20" Dia Penetration)	1.44E-1	1.09	7.90E-2
14" Dia Penetration	3.15E+0	1.23E+1	8.62E-1

6.8.2 Dose Rates Calculated Using QAD

Dose rates calculated via use of the QAD computer code are presented in the tables that follow. Dose rates are presented for the following situations:

- unshielded dose rates at 3 meters from the side of the reactor vessel along the reactor vessel active fuel midplane (AFM), at 3 meters above the reactor vessel along the reactor vessel centerline, and at 3 meters below the reactor vessel along the reactor vessel centerline
- unshielded dose rates at 1 meter from the side of the reactor vessel along the active fuel midplane, at 1 meter above the reactor vessel along the reactor vessel centerline, at 1 meter above the reactor vessel in the vicinity of the grid bar end pieces, at 1 meter above the reactor vessel - along the reactor vessel internal radius, and at 1 meter from the bottom of the reactor vessel along the reactor vessel centerline
- at contact i.e., 1" from the surface of the transport package at specified dose points
- at 2 meters from the surface of the transport package at specified dose points

The dose rates in these tables are summary dose rates i.e., they include the sum total of all the various source contributions, including the contributions due to radiation shine out the reactor vessel nozzle penetrations. Dose rates due to radiation shine out reactor vessel nozzle penetrations were determined without inclusion of a steel plug insertion in the nozzles. The resulting dose rates indicate that acceptance criteria dose rates can be obtained without the use of steel plugs for the nozzles. Specific contributions of each radiation source to the total dose point are presented in Attachment K to this calculation.

The contact and 2 meter shielded dose rates presented in the following tables are based on a transport container with a 4" thick iron top, a 3" thick iron bottom, and a 7" thick iron radial shield which extends from 16" above the active fuel region to 10" below the active fuel region. Dose rates are also dependent on there being cellular concrete with a minimum density of 30 (lbs/cu ft) in the reactor vessel and cellular concrete with a minimum density of 50 (lbs/cu-ft) in the area between the reactor vessel and the transport container. Contact and 2 meter dose rates at selected dose points are shown in Figure J-1 of Attachment J to this calculation.

QAD file names are identified in Attachment I to this calculation. QAD input and output files are included on the CD which accompanies this calculation (Attachment M).

Table 6-4. Unshielded 3 Meter Dose Rates from Reactor Vessel Filled with 30 lbs/cu ft LDCC

	Dose Rates (mrem/hr)		
	Side of Vessel	Below Vessel	Above Vessel
Total Dose Rate	5.44E+3	6.35E+0	3.48E+2

Table 6-5. Unshielded 1 Meter Dose Rates from Reactor Vessel Filled with 30 lbs/cu ft LDCC

	Dose Rate (mrem/hr)				
	Side of Vessel (Core Midplane)	Above Vessel (Along Centerline)	Above Vessel (Over Grid Bar Ends)	Above Vessel (at Inner Radius)	Below Vessel
Total Dose Rate	2.06E+4	7.93E+2	7.14E+2	6.62E+2	2.45E+1

Table 6-6. Shielded Contact Dose Rate to Transport Container

	Dose Rate (mrem/hr)					
	Active Fuel Midplane (AFM)	Side, Above 7" to 3" Radial Shield Transition i.e., 52" Above AFM	Side, Below 7" to 3" Radial Shield Transition i.e., 47" Below AFM	At 20" Dia Penetration El i.e., 81" Below AFM	Along RVTS Centerline Above Transport Container	Along RVTS Centerline Below Transport Container
Total Dose Rate	1.11E+1	5.28E+1	1.62E+1	1.39E+0	1.62E+1	9.27E-1

Table 6-7. Shielded 2 Meter Dose Rates Above & Below Transport Container

	Dose Rate (mrem/hr)	
	Along Centerline Above RVTS	Along Centerline Below RVTS
Total Dose Rate	8.67E+0	4.37E-1

Table 6-8. Shielded 2 Meter Dose Rates at Side of Transport Container

	Dose Rates (mrem/hr)											
	At AFC	Distance Below Active Fuel Midplane (AFM)					Distance Above Active Fuel Midplane (AFM)					
	----	37"	47"*	55"	69"	81"	37"	52"*	60"	74"	86"	110"
Total	4.39	3.24	3.09	2.90	2.48	2.23	4.22	6.35	6.62	7.18	7.31	4.97

* The distances of 47" and 52" below and above active fuel midplane are located just below and above the transition between the 7" to the 3" iron thickness along the radial side of the transport package.

It is observed from Table 6.6 that the maximum contact shielded dose rate occurs along the radial side of the transport package, above the transition point where the transport package radial shield thickness changes from 7" to 3" of iron. Tables 6-7 and 6-8 indicate that the maximum shielded 2 meter dose rate (8.7 mrem/hr) occurs above the RVTS, along the RVTS centerline. About 30% of the maximum shielded 2 meter dose rate is attributed to the grid bar end pieces source term (See Attachment K to this calculation). The maximum shielded 2 meter dose rate along the side

of the RVTS (7.3 mrem/hr) occurs several feet above the transition point where the transport package radial shield thickness changes from 7" to 3" of iron.

6.8.3 Dose Rates Due to Radiation Streaming through LDCC Fill Hole Plugs in Transfer Cask

ISOSHLD-PC calculated dose rates due to radiation streaming out the 3" diameter plugs in the transport container LDCC fill holes are presented in the table that follows. The dose rates are based on a 2" thick iron plug located in the LDCC fill holes and are 2 meters and 1" from the RV transport package outer surface, along the LDCC fill hole plug centerline.

The ISOSHLD-PC file name is identified in Attachment I to this calculation. ISOSHLD-PC input and output files are included on the CD which accompanies this calculation (Attachment M).

Table 6-9. Dose Rate due to Radiation Streaming out LDCC Fill Hole Plugs in RV Transport Container

Source Component	Contact Dose Rate (mrem/hr)	2 Meter Dose Rate (mrem/hr)
RV Insulation	9.15	3.82E-2
RV Wall/Cladding	1.91E+1	5.95E-2
Thermal Shield Retainer	3.76	1.09E-1
Thermal Shield Seal Weights	2.83	-----
Thermal Shield	2.95	-----
TOTAL	3.78E+1	2.07E-1

6.9 Evaluation for Radiation Shine through Penetration Nozzles N-12 & N-14

Reactor vessel nozzle penetrations N-12 and N-14 are 3" diameter penetrations that are closest to and above the grid bar end pieces (Design Input 3.12). The centerline for these penetrations is at 192" above the reactor vessel base line (Design Input 3.12). The reactor vessel wall is about 5.25" thick (Design Input 3.7). The top of the top guide plate is (1" + 14'2.75") = 171.75" above the base line (Design Input 3.6). The 6" long grid bar end pieces are in the vicinity of these penetrations (Design Inputs 3.6 and 3.12). The top of the 6" long grid bar end pieces are about (6" + 2.625" -.75") = 7.875" above the top guide plate when in the vertical position (Design Input 3.6). This puts the top of the grid bar end pieces at [(192"-1.5") – (171.75" + 7.875")] = 10.875" below the bottom of the penetrations. The distance (D) from the reactor vessel wall where direct shine from 10.875" below the penetration can pass through the penetration is determined by solving the following equation: [(5.25"/3") = (D/10.875")]. Solving for D gives a distance slightly greater than 19 inches. The minimum radial distance to the inner radius of the top guide plate annulus, in the vicinity of penetrations N-12 and N-14 is $[(30.25")^2 + (22\text{-}7/8")^2]^{0.5}$ = 37.9" [Drawing References 2.13.5 and 2.13.12]. The radial distance to the inside surface of the reactor vessel wall is about 53" (Design Input 3.7). Thus the distance from the inner radius of the top guide annulus to the reactor vessel inner wall surface is (53" – 37.9") = 15.1". Drawing 197E861 [Reference 2.13.12] shows the grid bar end pivot point. The pivot point is situated such that when the grid bar end pieces are in the vertical position, they are essentially over the top guide plate. Since the top of the vertical grid bar end pieces are 15.1" from the reactor vessel wall and since

shine from the top grid bar end pieces elevation through the penetration doesn't occur until a distance of 19" from the reactor vessel wall, there is no direct radiation shine from vertically oriented grid bars end pieces through the penetrations. This evaluation applies to grid bar end pieces in the immediate vicinity of the penetrations. Direct radiation from grid bar end pieces located on the opposite side of the reactor vessel from the penetrations can shine through the penetrations but the radiation is sufficiently shielded by the LDCC located in the reactor vessel.

When the grid bar end pieces are in the horizontal position, they are about 1.75" + 5.625" = 7.375" above the top guide plate (Design Input 3.6). This puts the top of the grid bar end pieces at [(192"-1.5") – (171.75" + 7.375")] = 11.375" below the bottom of the penetrations. The distance (D) from the reactor vessel wall where direct shine from 11.375" below the penetration can pass through the penetration is determined by solving the following equation: [(5.25"/3") = (D/11.375")]. Solving for D gives a distance of 19.9 inches. The minimum radial distance to the grid bar end (when grid bar end is in the horizontal position) is: $[(26.375")^2 + (23")^2]^{0.5}$ = 35". This distance is (53" – 35") = 18" from the inside surface of the reactor vessel wall. Since the farthest point on the horizontal grid bar end pieces is 18" from the reactor vessel wall and since shine from the top grid bar end pieces elevation through the penetration doesn't occur until a distance of 19.9" from the reactor vessel wall, there is no direct radiation shine from horizontally oriented grid bar end pieces through the penetrations. This evaluation applies to grid bar end pieces in the immediate vicinity of the penetrations. Direct radiation from grid bar end pieces located on the opposite side of the reactor vessel from the penetrations can shine through the penetrations but the radiation is sufficiently shielded by the LDCC located in the reactor vessel.

7.0 Summary and Conclusions

Dose rate results in Section 6.8 indicate that the acceptance criteria dose rate limits presented in Section 5 (i.e., 200 mrem/hr at contact to the RVTS and 10 mrem/hr at 2 meters from the RVTS) can be met when the following shielding design features are incorporated into the reactor vessel transport system.

- A cellular concrete with a minimum density of 30 pounds per cubic foot is used to fill void spaces in the reactor vessel. (Gamma dose rates are inversely proportional to shield material density thus a LDCC density greater than 30 pounds per cubic foot will result in smaller dose rates than those calculated in this evaluation.)
- A cellular concrete with a minimum density of 50 pounds per cubic foot is used to fill void spaces between the reactor vessel and the transport container. (Gamma dose rates are inversely proportional to shield material density thus a LDCC density greater than 50 pounds per cubic foot will result in smaller dose rates than those calculated in this evaluation.)
- The transport package includes iron shielding in the following minimum thicknesses:
  Top of transport package: 4" thick iron
  Bottom of transport package: 3" thick iron
  Radial side of transport package: 7" thick iron which extends from 16" above the top of the active fuel region to 10" below the bottom of the active fuel region. The remainder of the radial side of the transfer package is 3" thick iron.
- LDCC fill holes in the RV transport package are limited to 3" nominal diameter and contain an iron plug having a thickness no less than 1" thinner than the transport container wall.

Calculated dose rates were obtained which meet the acceptance criteria dose rate limits without inclusion of steel plugs in RV nozzle penetrations.

In addition to the aforementioned shielding design features, dose rates calculated in this evaluation are dependent on removal of the Neutron Windows as indicated in Design Input 3.2. The calculated dose rates also require that the Top Guide ("grid bars") be removed except for 6" and 10" length grid bar end pieces as indicated in Reference 2.18. If these components are not removed from the RVAI or if the aforementioned shielding design features are not incorporated into the RVTS, then the dose rates could exceed calculated values.

The maximum calculated dose rates from the RVTS are:

- 5.44E+3 mrem/hr at 3 meters from the unshielded side of the reactor vessel insulation
- 5.3E+1 mrem/hr at contact to the side of the shielded RVTS
- 8.7 mrem/hr at 2 meters above the top of the shielded RVTS
- 7.3 mrem/hr at 2 meters from the side of the shielded RVTS
- 2.1E+4 mrem/hr at 1 meter from the unshielded side of the reactor vessel insulation
- 7.93E+2 mrem/hr at 1 meter above the unshielded reactor vessel (along the reactor vessel centerline)

The shielded contact and 2 meter dose rates calculated at all dose points are within the acceptance dose rate criteria limits of 200 mrem/hr at contact and 10 mrem/hr at 2 meters. The maximum shielded dose rates from the RVTS are about 8.7 mrem/hr at 2 meters above the RVTS (along the RVTS centerline) and 7.3 mrem/hr at 2 meters from the side of the RVTS (about 4 feet above the active core region). The grid bar end pieces source term is the largest contributor to these maximum dose rates (i.e., this source term is responsible for at least 30% of the overall dose rates at these two dose points). Features incorporated into the QAD computer code model and into component radiation source term development (e.g., 15%

source term margin, conservative axial distribution of source terms, etc.,) result in calculated dose rates that are expected to be conservative.

Calculated dose rates at contact to and 2 meters from a plugged LDCC fill hole in the transport cask are presented in Section 6.8.3 to this calculation. The calculated dose rates are 37.8 mrem/hr at contact and 0.207 mrem/hr at 2 meters and are due to radiation streaming through the plugged LDCC fill hole. If the contact and 2 meter streaming dose rates are added to the maximum calculated dose rates presented above, the total contact and 2 meter dose rates are still well below the dose rate criteria.

The relationships between major radiation sources and reactor vessel nozzles were evaluated and it was determined that radiation shine from these sources through the nozzle penetrations would not be significant due to the distance between these sources and the penetrations and the attenuation provided by the in-vessel LDCC. The radiation dose rates streaming through these 14" diameter and 20" diameter nozzle penetrations were determined based on radiation source terms in the immediate vicinity of the penetrations. The nozzle penetration analyses indicates that the nozzle penetrations do not require steel plugs in order to attain dose rates outside the RVTS that are within acceptance criteria dose limits

8.0 Attachments

Attachment A: "Comparison of Activities and Part 61 Classification at Big Rock Point as of September 1, 2002", Attachment A, Corrected Total Activity (Ci), Reference 2.3, 1 page;

Attachment B: "Reactor Vessel Package Radioactivity and A_2 Fractions", Table 1-2, Reference 2.4, 1 page;

Attachment C: "PR-2 and PR-7 Series Specifications", DOSITEC, Inc., Sheet #8201-0293, 1 page;

Attachment D: "Radial and Axial Thermal Neutron Flux Distribution", Figures 2-2 and 2-3, Reference 2.2, 2 pages;

Attachment E: "Big Rock Point Reactor Pressure Vessel and Internals", Table 2-1, Reference 2.2, 1 page;

Attachment F: "Big Rock Point Reactor Pressure Vessel and Internals", Figure 2-1, Reference 2.2, 1 page;

Attachment G: Deleted;

Attachment H: "QAD Computer Model Description", 4 pages;

Attachment I: "ISOSHLD-PC/QAD Input/Output Files", 3 pages;

Attachment J: "Dose Point Locations and Source Term Information for QAD Model", 9 pages;

Attachment K: "QAD Dose Rate Results," 3 pages;

Attachment L: M. D. Papp (BNFL) to B. Slimp (S&L), "Big Rock Point Restoration Project - 5339 - Reactor Vessel Grid Bars and Neutron Windows," Letter No. BRP-2000-05-228 dated May 25, 2000, 1 page;

Attachment M: "CD for Calculation N-10525-020-0002", Rev 1, 1 page.

FINAL

Comparison of Activities and Part 61 Classification at Big Rock Point as of September 1, 2002
(Note Results Include Surface Contaminants)

Component Name	Original Total Activity (Ci)	Corrected Total Activity (Ci)	Percent Error	Original Co-60 Activity (Ci)	Corrected Co-60 Activity (Ci)	Percent Error	Original Part 61 Table 1 Fraction	Corrected Part 61 Table 1 Fraction	Percent Error	Original Part 61 Table 2 Fraction	Corrected Part 61 Table 2 Fraction	Percent Error
Greater Than Class C Waste												
Top Guide	4.84E+04	5.35E+04	9.63%	2.12E+04	2.42E+04	12.65%	8.55	9.78	12.59%	11.85	12.79	7.33%
GTCC Subtotals	4.84E+04	5.35E+04	9.63%	2.12E+04	2.42E+04	12.65%						
Low Level Radioactive Waste (LLRW)												
Steam Baffle	1.09E+01	1.08E+01	-0.94%	4.95E+00	4.92E+00	-0.62%	0.02	0.02	-0.05%	< 0.01	< 0.01	NA
Sparger	8.78E-01	8.60E-01	-2.16%	3.97E-01	3.96E-01	-0.25%	0.03	0.03	-0.02%	< 0.01	< 0.01	NA
Top Guide Plate	1.17E+03	1.14E+03	-2.95%	5.38E+02	5.40E+02	0.40%	0.41	0.41	0.32%	0.54	0.51	-6.94%
Seal Housing	1.45E+02	1.30E+02	-11.53%	6.60E+01	6.08E+01	-8.59%	0.04	0.04	-7.71%	0.05	0.05	-14.22%
Thermal Shield	7.45E+03	7.40E+03	-0.66%	3.83E+03	3.94E+03	2.68%	0.18	0.19	2.42%	0.20	0.19	-4.61%
Thermal Shield Retainer (6 Units)	1.18E+02	1.14E+02	-3.59%	5.94E+01	5.94E+01	-0.09%	0.07	0.07	-0.26%	0.08	0.07	-7.54%
Seal Weights (12 Units)	8.54E+02	1.10E+03	22.46%	5.89E+02	7.32E+02	18.49%	0.08	0.09	17.85%	0.04	0.05	28.74%
Neutron Windows (4 Units)	1.37E+04	1.34E+04	-2.24%	6.92E+03	6.99E+03	1.11%	2.09	2.11	0.93%	2.67	2.52	-6.06%
Core Support Plate	1.40E-01	1.40E-01	0.00%	1.36E-01	1.36E-01	0.00%	0.01	0.01	0.00%	< 0.01	< 0.01	NA
Inlet Diffuser (2 Units)	2.24E-02	2.24E-02	0.00%	2.18E-02	2.18E-02	0.00%	0.02	0.02	0.00%	< 0.01	< 0.01	NA
Inlet Baffle	1.63E-01	1.63E-01	0.00%	1.59E-01	1.59E-01	0.00%	0.01	0.01	0.00%	< 0.01	< 0.01	NA
LLRW Subtotals	2.35E+04	2.33E+04	-0.68%	1.20E+04	1.23E+04	2.61%						
Reactor Vessel Assembly												
Vessel Cladding	2.09E+02	2.45E+02	14.90%	1.30E+02	1.53E+02	14.76%	0.04	0.04	6.13%	0.01	0.01	15.15%
Reactor Vessel Wall	5.90E+02	1.11E+03	46.66%	1.52E+02	2.01E+02	24.42%	< 0.01	< 0.01	NA	< 0.01	< 0.01	NA
Vessel Insulation	5.21E+01	5.81E+01	10.26%	3.05E+01	3.45E+01	11.56%	< 0.01	< 0.01	NA	< 0.01	< 0.01	NA
Reactor Vessel Subtotals	8.51E+02	1.41E+03	39.63%	3.13E+02	3.88E+02	19.48%						
Package Totals	2.43E+04	2.47E+04	1.63%	1.23E+04	1.27E+04	3.12%						
Grand Total	7.27E+04	7.83E+04	7.10%	3.35E+04	3.69E+04	9.37%						

Table 1-2

Reactor Vessel Package Radioactivity and A_2 Fractions

Nuclide	Activity [1]		A_2 Limit (Curies)	Fraction A_2 Value
	Surface Contamination (Curies)	Activation (Curies)		
H-3	8.11E-02	6.55E+02	1080	6.07E-01
C-14	1.15E-01	2.18E+02	54.1	4.03E+00
Sb-125	1.67E+00	4.04E-01	24.3	8.53E-02
Ce-144	4.64E-02		5.41	8.58E-03
Mn-54	8.57E-02	2.16E+03	27.0	8.00E+01
Eu-152		2.54E+01	24.3	1.05E+00
Fe-55	2.77E+01	6.97E+05	1080	6.45E+02
Co-60	9.92E+01	1.15E+06	10.8	1.06E+05
Ni-59		9.53E+02	1080	8.82E-01
Ni-63	2.00E+01	1.57E+05	811	1.94E+02
Nb-94		3.29E+00	16.2	2.03E-01
Sr-90	9.24E-01		2.7	3.42E-01
Tc-99		7.06E-01	24.3	2.91E-02
Pu-238	8.35E-02		0.00541	1.54E+01
Pu-239/240	9.33E-02		0.00541	1.72E+01
Pu-241	5.07E+00		0.27	1.88E+01
Pu-242	4.70E-04		0.00541	8.69E-02
Cm-242	1.56E-05		0.27	5.78E-05
Cm-243	3.60E-02		0.00811	4.44E+00
Cm-244	3.41E-02		0.0108	3.16E+00
Am-241	1.10E-01		0.00541	2.03E+01
Total[2]	155.2	2.01E+06		1.08E+05

Notes: 1. Activity values have been decayed to 11/01/97.
2. Total does not include the contribution from U-234 (7.02E-03 Ci surface contamination and no activation), which is negligible.

[redacted]

Calculation No.: N-10525-020-0002
Revision 1
Attachment C
Page 1 of 1, FINAL

PR-7 Underwater Monitor

GENERAL DESCRIPTION

The Dositec Portable Remote Monitor PR-2 and PR-7 Series (PR-2L, PR-2M, PR-2H, PR-2XH and PR-7) are designed for remote underwater survey applicatons in gamma radiation fields up to 5, 500, 5,000, 50,000 and 50,000 R/hr, respectively. They can be used at distances up to 2000 feet; the Underwater Probe can be used to a depth of 150 feet.

Each of the PR-2 and PR-7 Series Monitors Consists of an Underwater Probe, a Meter Unit, and a Cable. The PR-2 and PR-7 Series Monitors employ energy-compensated solid-state detectors, which do not require high voltage. Power is supplied by a 9 volt Alkaline battery included.

Each of the PR-2 and PR-7 Series Monitors has an Analog Meter, which indicates radiation dose rates in three linear ranges for PR-2 and seven linear ranges for PR-7.

Calibration of the PR-2 and PR-7 Series Monitor is performed by simple adjustment of the potentiometers on the meter unit. Calibration is independent of cable length, up to 500 feet for PR-2 and 2000 feet for PR-7. For longer distances, recalibration is required.

DOSITEC, INC.
4 Avenue E
Hopkinton MA 01748-2211

PR-2 and PR-7 Series Specifications

DETECTION

Detector Type	Energy-compensated solid-state CdTe or Si-Detector
Radiation	X and Gamma Ray
Energy Response	±30% from 80 Kev to 3 Mev.
Dose Rate Range	
Model PR-2L	0-0.05, 0-0.5, and 0-5 R/hr
Model PR-2M	0-5, 0-50, and 0-500 R/hr
Model PR-2H	0-50, 0-500, and 0-5,000 R/hr
~~Model PR-2XH~~	~~0-500, 0-5,000 and 0-50,000 R/hr~~
Model PR-7	0-0.05, 0-0.5, 0-5, 0-50, 0-500, 0-5,000 and 0-50,000 R/hr
Accuracy	±15% for all ranges from 20% to 80% of full scale, calibrated with Cs137

UNDERWATER PROBE

Depth Capability	Up to 150 ft (45 m)
Housing	Nickel-plated Aluminum; 0.5" Wall, 0.2" bottom
Detector Location	10 mm from bottom center
Connector	Quick-connect to Cable
Dimensions	2.375" dia. x 5" length (60 mm dia. x 127 mm length)
Weight	1.5 lbs (0.7 kg)

METER UNIT

Display	Analog Meter
Response Time	5 to 20 seconds, 0 to 90% of final reading
Battery.	one 9 volt Alkaline battery with 100 hours life
Dimension(approx.)	
PR-2	2.50" x 4.50" x 1.75" (65 mm x 114 mm x 45 mm)
PR-7	2.62" x 4.75" x 1.50" (66 mm x 121 mm x 38 mm)
Weight	1 lb (0.5 kg).
Calibration	Potentiometer adjustment per range; independent of cable length. up to 500 ft for PR-2, and up to 2000 feet for PR-7

CABLE	50 ft (15 m), 2.5 lbs (1.1 kg) standard; up to 2000 ft (600 m) available

Rec'd.
07 FEB

All Specifications Subject to Change Without Notice

From Reference 2.2

neutron source equal to 2.49E+14 n/cm²-s, calculated by ORIGEN2 for an average Big Rock Point core.

The first radial model included the seal weights, which cover the majority of the thermal shield periphery.

The second radial model excluded the seal weights present in the first model to determine their effects on the flux in the surrounding components. A weighted average of the results from the first and second models was used to accurately determine the average flux in each radial component. The radial thermal flux results are shown in Figure 2-2.

The third radial model included neutron windows. This model was to determine the flux characteristics in the neutron windows and the impact on the flux levels in the components radially behind the neutron windows.

FIGURE 2-2
Big Rock Point Radial Thermal Flux Distribution

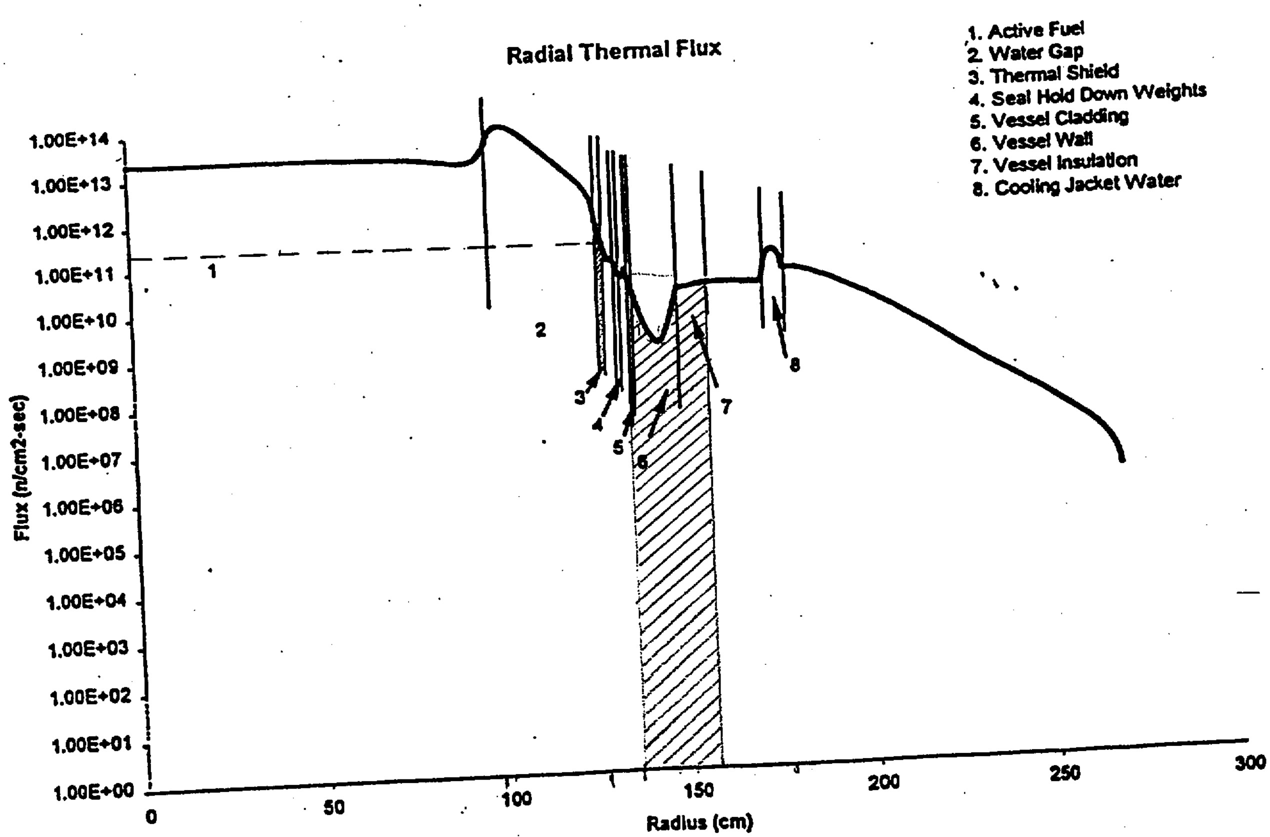
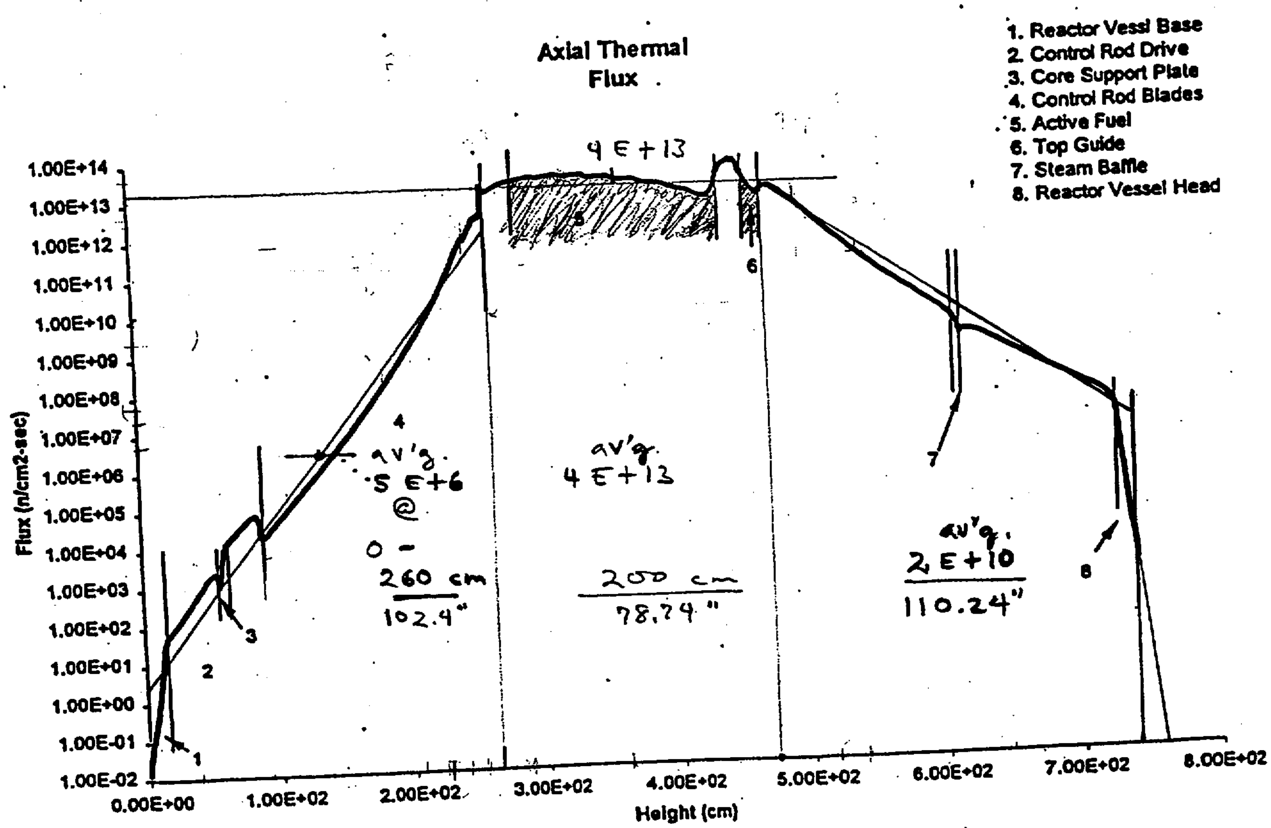


Figure 2-3

Big Rock Point Axial Thermal Flux Distribution



2.3 ORIGEN2 Point Neutron Activation Calculations

The ORIGEN2 computer code was used to calculate the activation and depletion of radionuclides in components exposed to neutrons. Neutron cross-sections from the BWRUS[7] library were used for the core activation, and neutron cross-sections from the thermal cross-section library were used for the thermal activation.

Based upon the ANISN spectral results, the core and thermal ORIGEN2 input fluxes were weighted to obtain a location-specific neutron spectrum for each component or region of interest. The thermal values were adjusted for the local area temperatures to reflect the reduced activation cross-sections at elevated temperatures.

TABLE 2-1

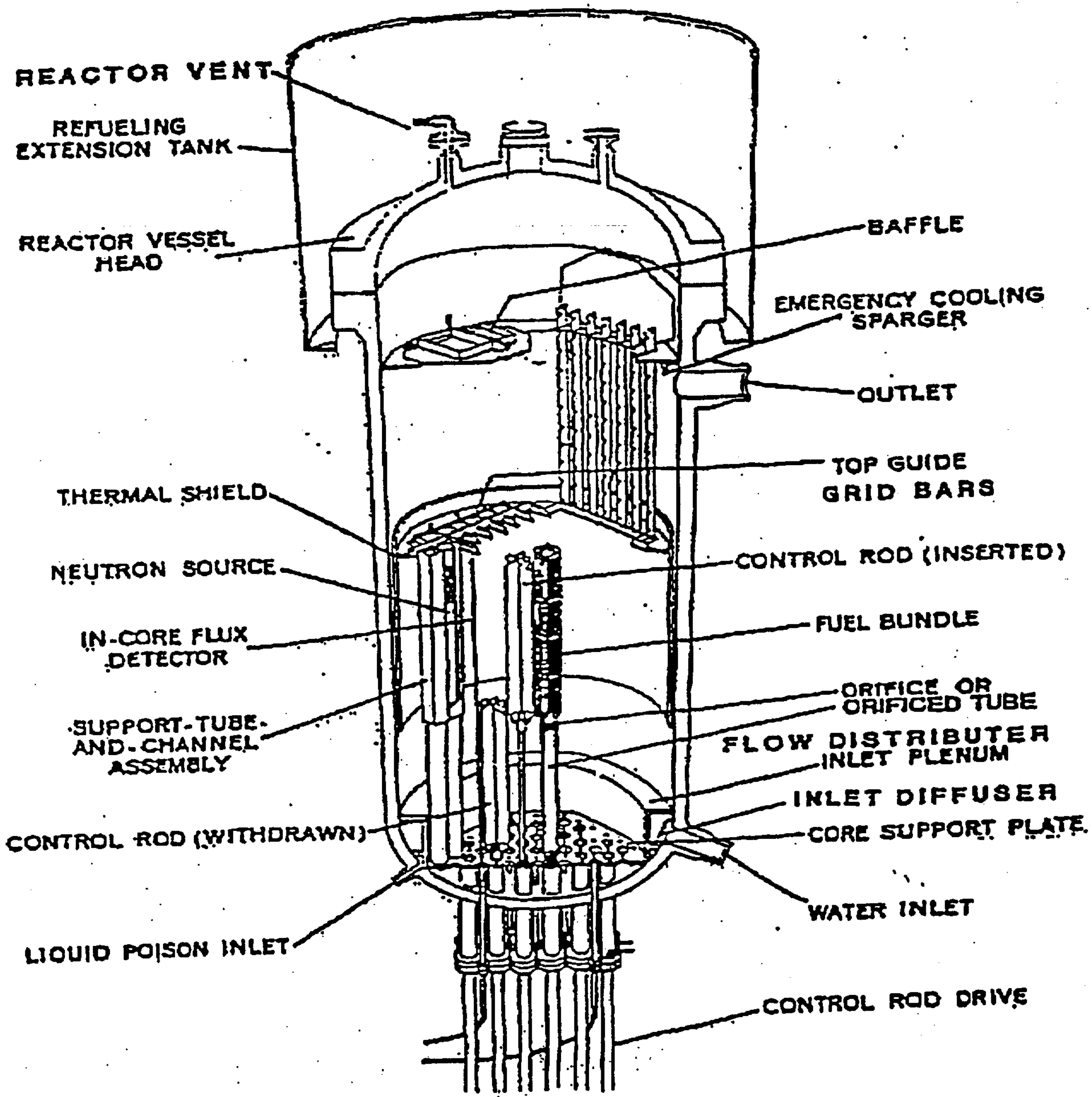
Big Rock Point Reactor Pressure Vessel and Internals

	Drawing Numbers	Length (inches)	Width / Thickness (inches)	Height (inches)	OD (inches)	Volume (ft^3)	Weight (lbs)
GTCC Material							
Top Guide	197E861					3.05E+00	1.53E+03
					Subtotal	**3.05E+00**	**1.53E+03**
LLRW Internals							
Steam Baffle	794E830		1.000		105.500	3.27E+00	1.64E+03
Sparger	M248 Sh46-1,41-2		2" SCH 40 PIPE		105.000	2.60E-01	1.30E+02
Top Guide Plate	197E118		1.000		99.188	1.63E+00	8.13E+02
Seal Housing	706E276			3.000	105.690	2.15E+00	1.07E+03
Thermal Shield	197E853		1.500	92.000	103.000	2.59E+01	1.30E+04
Thermal Shield Retainer (6 units)	237E956	10.000	0.750	14.130		1.09E+00	5.47E+02
Seal Weights (12 units)	706E276, 104R175	18.000	1.000	27.000		1.02E+01	5.08E+03
Neutron Windows (4 units)	107C353-9		0.718	72.000	6.450	2.46E+00	1.23E+03
Core Support Plate	762-D229, E201-809-3		1.500		79.250	5.08E+00	2.54E+03
Inlet Diffuser (2 units)	795E369	27.000	5.500	0.375		3.97E-01	1.98E+02
Inlet Baffle	795E421			36.000		4.64E+00	2.32E+03
					Subtotal	**5.71E+01**	**2.85E+04**
Reactor Vessel Assembly							
Reactor Vessel Assembly	197E853, 104R175, F 230-791-2			287.625	116.938	4.14E+02	2.03E+05
Reactor Vessel Insulation	795E369			290.625	122.500	2.59E+02	5.18E+03
					Subtotal	**6.73E+02**	**2.08E+05**

Other material types present in the vessel internals were also incorporated into the neutron transport models. These additional materials included zircaloy-2, inconel, ordinary concrete and vessel insulation of type 304 stainless steel at 20 lbs/ft^3.

From Reference 2.2

FIGURE 2-1
Big Rock Point Reactor Pressure Vessel and Internals



REACTOR VESSEL & INTERNALS

ATTACHMENT G

'DELETED'

ATTACHMENT H
QAD Computer Model Description

The model used in the QAD evaluation consists of 41 zones and 34 boundaries. (Although the QAD model contains 34 boundaries, only 33 of the boundaries were actually used in the final QAD runs. Boundary # 33 was used in intermediate runs to subdivide the transport vessel side shield thickness.). A zone represents a component or it represents space between or around components. The boundaries define the limitations for the zones. Figure H-1 shows a sectional view of the QAD geometry model used in this evaluation. The boundary dimensions are shown in Figure H-1. Boundary dimensions are given in inches and are either radial lengths or elevations in the 'Z' direction. The origin for the coordinate system is at the midplane of the active fuel, along the active fuel centerline. Boundary numbers are identified in Figure H-1 with the letter 'B' followed by the boundary number. Zones are identified by a number enclosed within a box. A description of the zones is provided in Table H-1 that follows.

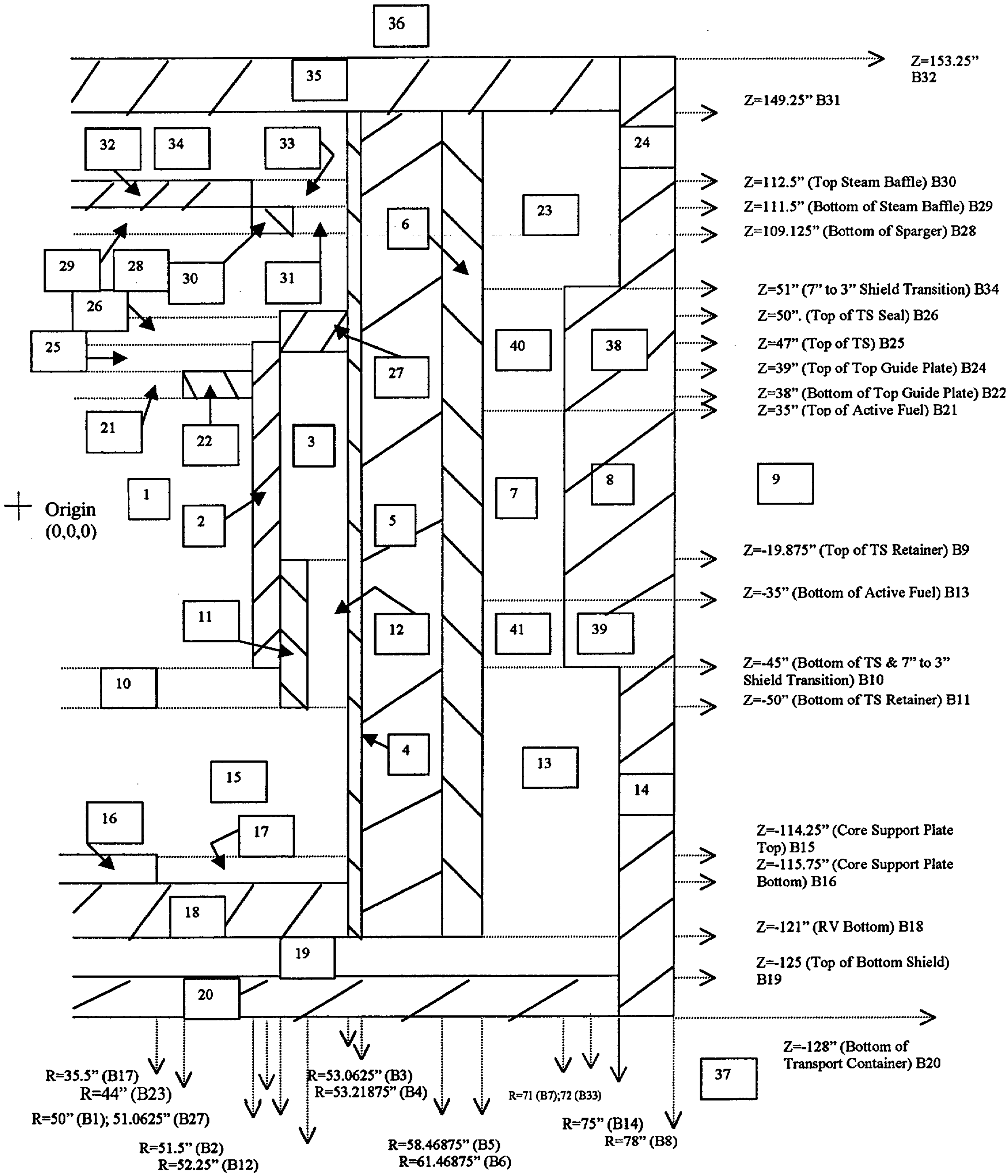
Note: Zones 38 through 41 for the QAD model which is used to determine unshielded and shielded dose rates associated with the grid bar end pieces differ from the zone description provided in Table H-1. For a description of these zones with respect to the grid bar end pieces QAD model, refer to Section 5.2.1 in the main body of this calculation.

Table H-1; QAD Model Zone Description

Zone Number	Zone Description	Model Discussion
1	Internal Vessel Cellular Concrete	Cylindrical area inside thermal shield filled with in-vessel cellular concrete
2	Thermal Shield	Annular area, 0 to 360 degrees
3	Internal Vessel Cellular Concrete	Annular area (0 to 360 degrees) between thermal shield and RV wall filled with in-vessel cellular concrete
4	Reactor Vessel Cladding	Annular area (0 to 360 degrees)
5	Reactor Vessel Wall	Annular area (0 to 360 degrees)
6	Reactor Vessel Insulation	Annular area (0 to 360 degrees)
7	Out-of-Vessel Cellular Concrete	Annular area (0 to 360 degrees)
8	Transport Container Side Shield	Annular area (0 to 360 degrees) for 7" thick iron shield band over the active fuel elevation
9	Air Outside & to Side of the Transport Container	Annular area (0 to 360 degrees)of air
10	Internal Vessel Cellular Concrete	Cylindrical geometry of internal vessel cellular concrete just below the thermal shield
11	Thermal Shield Retainer	10" wide by about 30" high segment of stainless steel. Extends from 84.4 degrees to 95.6 degrees.
12	Internal Vessel Cellular Concrete	Annular area (0 to 360 degrees) between TS retainer & RV wall
13	External Vessel Cellular Concrete	Annular area (0 to 360 degrees) between RV wall & transport container
14	Transport Container Side Shield	Annular area (0 to 360 degrees) for 3" thick iron shield band below the active fuel elevation
15	Internal Vessel Cellular Concrete	Cylindrical geometry of internal vessel cellular concrete just below the TS retainer
16	Core Support Plate	Cylindrical geometry near bottom of RV

Table H-1; QAD Model Zone Description		
Zone Number	Zone Description	Model Discussion
17	Internal Vessel Cellular Concrete	Annular area (0 to 360 degrees) adjacent to core support plate
18	Reactor Vessel Bottom	Cylindrical geometry. The model for reactor vessel bottom is conservatively located directly below the core support plate.
19	External Vessel Cellular Concrete	Cylindrical geometry directly below RV bottom
20	Bottom Shield of Transport Container	Cylindrical geometry, 3" thick iron
21	Internal Vessel Cellular Concrete	Cylindrical geometry adjacent to top guide plate
22	Top Guide Plate	Annular area (0 to 360 degrees) near top of thermal shield
23	External Vessel Cellular Concrete	Annular area (0 to 360 degrees) between RV wall & 3" thick transport container section above active fuel region
24	Transport Container 3" Thick Side Wall Above Active Fuel Region	Annular area (0 to 360 degrees)
25	Internal Vessel Cellular Concrete	Cylindrical geometry above top guide plate
26	Internal Vessel Cellular Concrete	Cylindrical geometry above thermal shield
27	Thermal Shield Seal	Annular area (0 to 360 degrees) above thermal shield
28	Internal Vessel Cellular Concrete	Cylindrical geometry above thermal shield seal
29	Internal Vessel Cellular Concrete	Cylindrical geometry adjacent to sparger
30	Sparger	Annular area (0 to 360 degrees) under steam baffle near top of RV
31	Internal Vessel Cellular Concrete	Annular geometry between sparger & RV wall
32	Steam Baffle	Cylindrical geometry
33	Internal Vessel Cellular Concrete	Annular geometry between steam baffle & RV wall
34	Internal Vessel Cellular Concrete	Cylindrical geometry above steam baffle
35	Transport Container Top Shield Plate	Cylindrical geometry. Top shield plate is at RV flange elevation
36	Air	Above transport container
37	Air	Below transport container
38	Transport Container Side Wall Shield Section	7" thick iron side wall segment which extends 16" up from top of active fuel region
39	Transport Container Side Wall Shield Section	7" thick iron side wall segment which extends 10" down from bottom of active fuel region
40	External Vessel Cellular Concrete	Annular geometry between RV wall & Zone 38
41	External Vessel Cellular Concrete	Annular geometry between RV wall & Zone 39

Figure H-1: QAD Geometry Model



Description of Radial Distances used in QAD Model:

Radial distances are shown in Figure H-1. A description of each radial distance follows. Note that a description of axial distances is provided in Figure H-1.

Table H-2; Description of Radial Distances	
Radial Distance (Inches)	Description
35.5	Outer radius of core support plate
44.	Inner radius for top guide plate (Outer radius is at thermal shield inner radius)
50.	Inner radius of thermal shield
51.0625"	Inner radius for thermal shield seal at top of thermal shield (Outer radius is at reactor vessel wall cladding inner radius)
51.5	Thermal shield outer radius
52.25	Outer radius for thermal shield retention plates (Inner radius is equal to thermal shield outer radius)
53.0625	Inner radius of reactor vessel wall cladding
53.21875	Outer radius of reactor vessel wall cladding (Also inner radius for reactor vessel wall)
58.46875	Outer radius of reactor vessel wall (Also inner radius of reactor vessel insulation)
61.46875	Outer radius of reactor vessel insulation
71.	Inner radius of 7" thick section of reactor vessel transport package iron shield
75"	Inner radius of 3" thick section of reactor vessel transport package iron shield
78"	Outer radius of reactor vessel transport package

FINAL

ATTACHMENT I
ISOSHLD-PC/QAD Input/Output Files

A listing of the directory which contains the ISOSHLD-PC/QAD executable files follows. The PC used to run the code for this evaluation is also identified.

```
 Volume in drive C is PC5597
 Volume Serial Number is 3311-10DC
 Directory of C:\BigRock\10525-Rev1\RUNTEx\ex

.              <DIR>        11-01-00  7:29a .
..             <DIR>        11-01-00  7:29a ..
ISDRTP   LIB       232,018  08-13-91 11:39a ISDRTP.LIB
ISOSHLD  EXE     1,056,526  02-29-00  2:17p ISOSHLD.EXE
2 file(s)      1,288,544 bytes
        2 dir(s)      78,741,504 bytes free
```

A list and description of the ISOSHLD-PC and QAD output files is contained in the table that follows. Note that the output files contain a complete listing of the input data. As such only the output files are included in the table. The input files have the same name as the output files except they end in 'dat' instead of 'txt'.

All of the ISOSHLD-PC and QAD input and output files that were used in this evaluation are included in the CD which accompanies this calculation.

ISOSHLD-PC/QAD Files		
Number	File Name	Discription
ISOSHLD-PC Run		
1	NOZDS.txt	Determines 3 meter unshielded and shielded contact/2 meter nozzle dose rates from 14" diameter and 20" diameter nozzle penetrations
2	PLUGS.txt	Determines dose rates due to radiation streaming out LDCC fill hole plugs in transport container side shield. Contact to and 2 meter dose rates are determined.
QAD Runs		
1	RV73AC.txt	Determines dose rates due to reactor vessel source at active core elevation
2	RV73A1.txt	Determines dose rates due to reactor vessel source from top of active core to top of thermal shield
3	RV73A2.txt	Determines dose rates due to reactor vessel source from top of thermal shield to 20" above thermal shield

ISOSHLD-PC/QAD Files		
Number	File Name	Discription
4	RV73A3.txt	Determines dose rates due to reactor vessel source from 20" above thermal shield to steam baffle
5	RV73A4.txt	Determines dose rates due to reactor vessel source above the steam baffle
6	RV73B1.txt	Determines dose rates due to reactor vessel source from bottom of active core to 4" below
7	RV73B2.txt	Determines dose rates due to reactor vessel source from 4" below active core to bottom of thermal shield retainer
8	RV73B3.txt	Determines dose rates due to reactor vessel source from bottom of thermal shield retainer to 20" below thermal shield retainer
9	RV73B4.txt	Determines dose rates due to reactor vessel source at distances greater than 20" below thermal shield retainer
10	IN73AC.txt	Determines dose rates due to reactor vessel insulation source at active core elevation
11	IN73A1.txt	Determines dose rates due to reactor vessel source from top of active core to top of thermal shield
12	IN73A2.txt	Determines dose rates due to reactor vessel insulation source from top of thermal shield to 20" above thermal shield
13	IN73A3.txt	Determines dose rates due to reactor vessel insulation source from 20" above thermal shield to steam baffle
14	IN73A4.txt	Determines dose rates due to reactor vessel insulation source above the steam baffle
15	IN73B1.txt	Determines dose rates due to reactor vessel insulation source from bottom of active core to 4" below
16	IN73B2.txt	Determines dose rates due to reactor vessel insulation source from 4" below active core to bottom of thermal shield retainer
17	IN73B3.txt	Determines dose rates due to reactor vessel insulation source from bottom of thermal shield retainer to 20" below thermal shield retainer
18	IN73B4.txt	Determines dose rates due to reactor vessel insulation source at distances greater than 20" below thermal shield retainer
19	TS73AC.txt	Determines dose rates due to thermal shield source at active core elevation
20	TS73AB.txt	Determines dose rates due to thermal shield source above active core elevation
21	TS73B1.txt	Determines dose rates due to thermal shield source from bottom of active core to 4" below

ISOSHLD-PC/QAD Files		
Number	File Name	Discription
22	TS73B2.txt	Determines dose rates due to thermal shield source from 4" below active core to bottom of thermal shield
23	TSR73AC.txt	Determines dose rates due to thermal shield retainer source at active core elevation
24	TSR73B.txt	Determines dose rates due to thermal shield retainer source belowt active core elevation
25	SW73AC.txt	Determines dose rates due to thermal shield seal weights source at active core elevation
26	SW73AB.txt	Determines dose rates due to thermal shield seal weights source above active core elevation
27	SW73B.txt	Determines dose rates due to thermal shield seal weights source below active core elevation
28	SB73.txt	Determines dose rates due to steam baffle source which is located above active core elevation
29	SP73.txt	Determines dose rates due to sparger source which is located above active core elevation
30	TGP73.txt	Determines dose rates due to top guide plate source which is located above active core elevation
31	TSS73.txt	Determines dose rates due to thermal shield seal housing source which is located above active core elevation
32	CSP73.txt	Determines dose rates due to core support plate source which is located below active core elevation. Source for inlet baffle and inlet diffuser is also included.
33	GBAR73.txt	Determines dose rates due to grid bar end pieces which are located directly above the top guide plate.

FINAL

Dose Point Locations & Coordinates

Dose Point #	Location Description	Coordinates
	Unshielded 3 Meter Distance	
1	Side of RV wall at active core midplane	r = 180.", phi = 90 degrees and z = 0."
2	Above RV at RV Centerline	x =0., y = 0., z = 267.0"
3	Below RV at RV Centerline	x =0., y = 0., z = -239.0"
	Shielded: Side of Vessel	
1	1" from Container Wall, at active core midplane	r = 79", phi = 90 degrees and z = 0."
2	2M from Container Wall, at active core midplane	r = 156.7", phi = 90 degrees and z = 0."
3	2M from Container Wall, bottom of active core	r = 156.7", phi = 90 degrees and z = -37."
4	2M from Container Wall, top of active core	r = 156.7", phi = 90 degrees and z = 37."
5	1" from Container Wall, 7" to 3" shield transition above active core	r = 79.", phi = 90 degrees and z = 52."
6	2M from Container Wall, 7" to 3" shield transition above active core	r = 156.7", phi = 90 degrees and z = 52."
7	1" from Container Wall, 7" to 3" shield transition below active core	r = 79", phi = 90 degrees and z = -47."
8	2M from Container Wall, 7" to 3" shield transition below active core	r = 156.7", phi = 90 degrees and z = -47."
9	2M from Container Wall, 60" above active core centerline	r = 156.7", phi = 90 degrees and z = 60."
10	2M from Container Wall, 74" above active core centerline	r = 156.7", phi = 90 degrees and z = 74."
11	2M from Container Wall, 86" above active core centerline	r = 156.7", phi = 90 degrees and z = 86."
12	2M from Container Wall, at steam baffle elevation above top of active core	r = 156.7", phi = 90 degrees and z = 110."
13	2M from Container Wall, 55" below active core centerline	r = 156.7", phi = 90 degrees and z = -55."
14	2M from Container Wall, 69" below active core centerline	r = 156.7", phi = 90 degrees and z = -69."
15	2M from Container Wall, 81" below active core centerline	r = 156.7", phi = 90 degrees and z = -81."
16	1" from Container Wall, 81" below active core centerline, pen El	r = 79.", phi = 90 degrees and z = -81."
	Shielded: Bottom of Vessel	
1	1" below Container along RV core CL	x, y, z = 0, 0, -129.
2	2M below Container along RV core CL	x, y, z = 0, 0, -206.7
3	1" below Container along RV inside diameter	x, y, z = 51., 0, -129.
4	2M below Container along RV inside diameter	x, y, z = 51., 0, -206.7
	Shielded: Top of Vessel	
1	1" above Container along RV core CL	x, y, z = 0, 0, 154.25
2	2M above Container along RV core CL	x, y, z = 0, 0, 232.
3	1" above Container along RV inside diameter	x, y, z =51., 0, 154.25
4	2M above Container along RV inside diameter	x, y, z = 51., 0, 232.
	Unshielded 1 Meter Distance	
1	Side of RV wall at active core midplane	r = 100.84.", phi = 90 degrees and z = 0."
2	Above RV at RV Centerline	x =0., y = 0., z = 188.62"
3	Above RV, Above Grid Bar End Pieces	r = 47.", phi = 90 degrees and z = 188.62"
4	Above RV, At RV Inner Radius	r = 52.", phi = 90 degrees and z = 188.62"
5	Below RV at RV Centerline	x =0., y = 0., z = -160.37"

FIGURE WITHHELD UNDER 10 CFR 2.390

Radiation Source Term Activity Determination

Active Core Region (Inventory Homogenized Over Components in Active Core Region)

			Activity (Curies)					
Nuclide	**QAD Nuclide ID #**	**Source Fraction**	**Thermal Shield**	**RV Cladding****	**RV Wall******	**Total (RV Wall & Clad)**	**RV Insulation**	**Thermal Shield Retainer*****
Measured Activity (Curies)			7.40E+03	2.45E+02	1.11E+03		5.81E+01	1.14E+02
Adjusted Activity (Curies)*			8.51E+03	9.39E+00			6.68E+01	2.19E+01
H-3	**451**	3.26E-04	2.77E+00	3.06E-03	4.16E-01	4.19E-01	2.18E-02	7.12E-03
C-14	**452**	1.08E-04	2.31E+02	2.54E-01	3.46E+01	3.49E+01	1.81E+00	5.92E-01
Sb-125	**269**	2.01E-07	1.71E-03	1.89E-06	2.56E-04	2.58E-04	1.34E-05	4.39E-06
Mn-54	**473**	1.07E-03	9.11E+00	1.00E-02	1.37E+00	1.38E+00	7.15E-02	2.34E-02
Eu-152	**408**	1.26E-05	1.07E-01	1.18E-04	1.61E-02	1.62E-02	8.42E-04	2.75E-04
Fe-55	**474**	3.47E-01	2.96E+03	3.27E+00	9.09E+02	9.13E+02	2.32E+01	7.60E+00
Co-60	**481**	5.72E-01	4.87E+03	5.37E+00	2.66E+02	2.71E+02	3.82E+01	1.25E+01
Ni-59	**None**	4.74E-04	4.03E+00	4.45E-03	6.06E-01	6.11E-01	3.17E-02	1.04E-02
Ni-63	**None**	7.81E-02	6.65E+02	7.33E-01	9.98E+01	1.01E+02	5.22E+00	1.71E+00
Nb-94	**111**	1.64E-06	1.40E-02	1.54E-05	2.09E-03	2.11E-03	1.10E-04	3.58E-05
Tc-99	**141**	3.51E-07	2.99E-03	3.30E-06	4.49E-04	4.52E-04	2.35E-05	7.67E-06

* Adjusted activity is 15% greater than measured activity as indicated in Section 6.1.
** Adjusted cladding activity includes a divisor of 30 to account for attenuation thru RV Wall per Design Input 3.13.
*** Adjusted thermal shield retainer activity includes a divisor of 6 to account for activity in 1 of 6 retainers.
**** Refer to Table 3-1.1 in Calculation body for nuclide distribution in the RV Wall.

Thermal Shield Seal Weights. There are 12 Seal Weights.
Seal Weights are staggered and can extend from bottom of thermal shield to 4" above active core region.
Seal weight adjusted activity is 1.15 times measured activity x (6/12) x (segment length/81").
Factor of 6/12 is to account for activity in 6 seal weights. 81" represents length range of seal weights.
Segment length is 81" in active core region, 4" above and 10" below active core region.

Thermal Shield Seal Weights & Portion of Retainer Below Active Core Region

			Activity (Curies) in 6 Seal Weights			Retainer
Nuclide	QAD Nuclide ID #	Source Fraction	Active Core Region	Above Active Core Region (4/81)	Below Active Core Region (10/81)	Below Active Core Region (15/30)*
Measured Activity (Curies)			1.10E+03	1.10E+03	1.10E+03	1.14E+02
Adjusted Activity in 6 Wts (Curies)			6.33E+02	6.33E+02	3.16E+02	5.46E+00
H-3	451	3.26E-04	2.06E-01	1.02E-02	1.27E-02	1.78E-03
C-14	452	1.08E-04	1.71E+01	8.46E-01	1.06E+00	4.27E-01
Sb-125	269	2.01E-07	1.27E-04	6.28E-06	7.85E-06	1.10E-06
Mn-54	473	1.07E-03	6.77E-01	3.34E-02	4.18E-02	5.84E-03
Eu-152	408	1.26E-05	7.97E-03	3.94E-04	4.92E-04	6.88E-05
Fe-55	474	3.47E-01	2.20E+02	1.09E+01	1.36E+01	1.90E+00
Co-60	481	5.72E-01	3.62E+02	1.79E+01	2.23E+01	3.12E+00
Ni-59	None	4.74E-04	3.00E-01	1.48E-02	1.85E-02	2.59E-03
Ni-63	None	7.81E-02	4.94E+01	2.44E+00	3.05E+00	4.27E-01
Nb-94	111	1.64E-06	1.04E-03	5.12E-05	6.40E-05	8.96E-06
Tc-99	141	3.51E-07	2.22E-04	1.10E-05	1.37E-05	1.92E-06

*Adjusted activity in retainer is 1.15 x measured activity x (15/30) x 0.5 x (1/6). (15/30) is portion below active core, 0.5 accounts for reduced activation due to flux reduction, (1/6) refers to source in 1 of 6 retainers.

Activity in Other Components (that are either above or below the active core region):

			Activity (Curies)					
Nuclide	**QAD Nuclide ID #**	**Source Fraction**	**Steam Baffle**	**Sparger**	**Top Guide Plate**	**Thermal Shield Seal Housing**	**Core Support Plate***	**Grid Bar End Pieces**
Measured Activity (Curies)			1.08E+01	8.60E-01	1.14E+03	1.30E+02	2.00E+00	1.80E+03
Adjusted Activity (Curies)			1.24E+01	9.89E-01	1.31E+03	1.50E+02	2.30E+00	1.80E+03
H-3	**451**	3.26E-04	4.05E-03	3.22E-04	4.27E-01	4.87E-02	7.50E-04	5.87E-01
C-14	**452**	1.08E-04	3.36E-01	2.68E-02	3.55E+01	4.05E+00	6.23E-02	4.88E+01
Sb-125	**269**	2.01E-07	2.50E-06	1.99E-07	2.64E-04	3.00E-05	4.62E-07	3.62E-04
Mn-54	**473**	1.07E-03	1.33E-02	1.06E-03	1.40E+00	1.60E-01	2.46E-03	1.93E+00
Eu-152	**408**	1.26E-05	1.56E-04	1.25E-05	1.65E-02	1.88E-03	2.90E-05	2.27E-02
Fe-55	**474**	3.47E-01	4.32E+00	3.44E-01	4.56E+02	5.20E+01	8.00E-01	6.26E+02
Co-60	**481**	5.72E-01	7.10E+00	5.66E-01	7.50E+02	8.55E+01	1.32E+00	1.03E+03
Ni-59	**None**	4.74E-04	5.89E-03	4.69E-04	6.21E-01	7.09E-02	1.09E-03	8.53E-01
Ni-63	**None**	7.81E-02	9.70E-01	7.72E-02	1.02E+02	1.17E+01	1.80E-01	1.41E+02
Nb-94	**111**	1.64E-06	2.04E-05	1.62E-06	2.15E-03	2.45E-04	3.77E-06	2.95E-03
Tc-99	**141**	3.51E-07	4.36E-06	3.47E-07	4.60E-04	5.25E-05	8.07E-07	6.32E-04

Adjusted activity is 1.15 times measured activity except for the Grid Bar End Pieces source term.
* Adjusted Core Support Plate Activity includes the sum of (1.4E-01 curies from core support, 2.24E-2 curies for inlet diffuser, and 1.63E-01 curies for inlet baffle). This is 0.3254 curies. An additional 1 curie is added to account for deposited crud and activation of the RV wall in the vicinity. The total activity is rounded up to 2 curies.

Reactor Vessel Wall and Insulation Below Active Core Region

			Activity (Curies)																
			4" Below Active Core				4" Below AC to Bottom of TS Retainer				To 20" Below TS Retainer				Greater than 20" Below TS Retainer			
Nuclide	QAD Nuclide ID #	Source Fraction	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation
Measured Activity (Curies)			2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01
Adjusted Activity (Curies)*			2.68E-01	3.65E+01		1.91E+00	1.24E-01	1.68E+01		8.82E-01	7.25E-03	9.85E-01		5.15E-02	4.58E-04	6.23E-02		3.26E-03
H-3	451	3.26E-04	8.75E-05	1.19E-02	1.20E-02	6.22E-04	4.04E-05	5.50E-03	5.54E-03	2.88E-04	2.36E-06	3.21E-04	3.24E-04	1.68E-05	1.49E-07	2.03E-05	2.05E-05	1.06E-06
C-14	452	1.08E-04	7.27E-03	9.89E-01	9.96E-01	5.17E-02	3.36E-03	4.57E-01	4.60E-01	2.39E-02	1.96E-04	2.67E-02	2.69E-02	1.40E-03	1.24E-05	1.69E-03	1.70E-03	8.83E-05
Sb-125	269	2.01E-07	5.39E-08	7.33E-06	7.38E-06	3.84E-07	2.49E-08	3.39E-06	3.41E-06	1.77E-07	1.46E-09	1.98E-07	1.99E-07	1.04E-08	9.21E-11	1.25E-08	1.26E-08	6.56E-10
Mn-54	473	1.07E-03	2.87E-04	3.91E-02	3.94E-02	2.04E-03	1.33E-04	1.81E-02	1.82E-02	9.44E-04	7.75E-06	1.06E-03	1.06E-03	5.52E-05	4.91E-07	6.68E-05	6.73E-05	3.49E-06
Eu-152	408	1.26E-05	3.38E-06	4.60E-04	4.63E-04	2.41E-05	1.56E-06	2.13E-04	2.14E-04	1.11E-05	9.13E-08	1.24E-05	1.25E-05	6.49E-07	5.78E-09	7.86E-07	7.92E-07	4.11E-08
Fe-55	474	3.47E-01	9.33E-02	2.60E+01	2.61E+01	6.64E-01	4.31E-02	1.20E+01	1.20E+01	3.07E-01	2.52E-03	7.02E-01	7.04E-01	1.79E-02	1.59E-04	4.44E-02	4.45E-02	1.13E-03
Co-60	481	5.72E-01	1.53E-01	7.59E+00	7.74E+00	1.09E+00	7.09E-02	3.51E+00	3.58E+00	5.04E-01	4.14E-03	2.05E-01	2.09E-01	2.95E-02	2.62E-04	1.30E-02	1.32E-02	1.87E-03
Ni-59	None	4.74E-04	1.27E-04	1.73E-02	1.74E-02	9.05E-04	5.88E-05	8.00E-03	8.06E-03	4.18E-04	3.43E-06	4.68E-04	4.71E-04	2.44E-05	2.17E-07	2.96E-05	2.98E-05	1.55E-06
Ni-63	None	7.81E-02	2.10E-02	2.85E+00	2.87E+00	1.49E-01	9.68E-03	1.32E+00	1.33E+00	6.89E-02	5.66E-04	7.70E-02	7.76E-02	4.03E-03	3.58E-05	4.87E-03	4.91E-03	2.55E-04
Nb-94	111	1.64E-06	4.40E-07	5.98E-05	6.02E-05	3.13E-06	2.03E-07	2.76E-05	2.78E-05	1.45E-06	1.19E-08	1.61E-06	1.63E-06	8.45E-08	7.52E-10	1.02E-07	1.03E-07	5.35E-09
Tc-99	141	3.51E-07	9.42E-08	1.28E-05	1.29E-05	6.70E-07	4.35E-08	5.92E-06	5.96E-06	3.10E-07	2.54E-09	3.46E-07	3.49E-07	1.81E-08	1.61E-10	2.19E-08	2.21E-08	1.14E-09

*Adjusted Activity is: 1.15 times measured activity times: 0.5 * 4/70 for areas from active core bottom to 4" below active core. 4/70 is component length over active core length. 0.5 factor per Section 6.4.
Adjusted Activity is: 1.15 times measured activity times: 11/70 * 0.084 for areas from 4" Below AC to bottom of TS Retainer. Factor of 0.084 per Section 6.4. Factor of 11/70 is component to active core length.
Adjusted Activity is: 1.15 times measured activity times: 20/70 * 2.7E-3 for areas between bottom of the TS retainer and 20" below TS retainer. Factor of 2.7E-3 is per Section 6.4. Factor of 20/70 is component to active core length.
Adjusted Activity is: 1.15 times measured activity times: 51/70 * 6.7E-5 for areas greater than 20" below TS Retainer. Factor of 6.7E-5 is per Section 6.4. Factor of 51/70 is component length over active core length.
In addition to the source term factors described above, RV clad has additional divisor of 30 per Design Input 3.13.

Thermal Shield Activity Above & Below Active Core Region

			Activity (Curies)			
Nuclide	QAD Nuclide ID #	Source Fraction	Active Core Region	Above Active Core Region (12/70)	4" Below Active Core Region (4/70)	4" to 10" Below Active Core Region (6/70)
Measured Activity (Curies)			7.40E+03	7.40E+03	7.40E+03	7.40E+03
Adjusted Activity (Curies)			8.51E+03	8.42E+02	2.43E+02	6.13E+01
H-3	451	3.26E-04	2.77E+00	2.74E-01	7.93E-02	2.00E-02
C-14	452	1.08E-04	2.31E+02	2.28E+01	6.59E+00	1.66E+00
Sb-125	269	2.01E-07	1.71E-03	1.69E-04	4.89E-05	1.23E-05
Mn-54	473	1.07E-03	9.11E+00	9.01E-01	2.60E-01	6.56E-02
Eu-152	408	1.26E-05	1.07E-01	1.06E-02	3.06E-03	7.72E-04
Fe-55	474	3.47E-01	2.96E+03	2.93E+02	8.45E+01	2.13E+01
Co-60	481	5.72E-01	4.87E+03	4.81E+02	1.39E+02	3.50E+01
Ni-59	None	4.74E-04	4.03E+00	3.99E-01	1.15E-01	2.90E-02
Ni-63	None	7.81E-02	6.65E+02	6.57E+01	1.90E+01	4.79E+00
Nb-94	111	1.64E-06	1.40E-02	1.38E-03	3.99E-04	1.00E-04
Tc-99	141	3.51E-07	2.99E-03	2.95E-04	8.53E-05	2.15E-05

Adjusted activity is measured activity x 1.15 x component length ratio x source adjustment factors.
Component length ratios are 1/1 at active core (AC), (12/70) above AC, (4/70) and (6/70) below AC.
Source adjustment factors are 0.577 above AC, 0.5 to 4" below AC & 0.084 at 4" to 10" below.

Activity (Curies) for Components that Shine thru Nozzles

			Bottom Nozzle		Top Nozzle		
Nuclide	QAD Nuclide ID #	Source Fraction	Diffuser	Inlet Baffle	Steam Baffle	Sparger	Total
Measured Activity (Curies)			2.24E-02	1.63E-01	1.08E+01	8.60E-01	
Adjusted Activity (Curies)			1.29E-02	1.12E-02	2.73E-01	0.130468787	4.04E-01
H-3	451	3.26E-04	4.20E-06	3.66E-06			1.32E-04
C-14	452	1.08E-04	3.49E-04	3.04E-04			1.09E-02
Sb-125	269	2.01E-07	2.59E-09	2.26E-09			8.11E-08
Mn-54	473	1.07E-03	1.38E-05	1.20E-05			4.32E-04
Eu-152	408	1.26E-05	1.62E-07	1.41E-07			5.09E-06
Fe-55	474	3.47E-01	4.48E-03	3.90E-03			1.40E-01
Co-60	481	5.72E-01	7.37E-03	6.42E-03			2.31E-01
Ni-59	None	4.74E-04	6.11E-06	5.32E-06			1.91E-04
Ni-63	None	7.81E-02	1.01E-03	8.77E-04			3.15E-02
Nb-94	111	1.64E-06	2.11E-08	1.84E-08			6.62E-07
Tc-99	141	3.51E-07	4.52E-09	3.94E-09			1.42E-07

Adjusted activity is measured activity times 1.15. Additional factors are used for inlet baffle, steam baffle & sparger. The additional factors account for fraction of source that shines thru penetration (5.98E-2 for inlet baffle; 0.022 for steam baffle; & 0.132 for sparger). Source fractions are determined in Section 6.6.

FINAL

Reactor Vessel Wall and Insulation Above Active Core Region

			Activity (Curies)															
			Active Core to Top of TS				Top of TS to 20" Above TS				20" Above TS to Baffle				From Steam Baffle & Above			
Nuclide	QAD Nuclide ID #	Source Fraction	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation	RV Cladding	RV Wall	Total (Wall and Cladding)	RV Insulation
Measured Activity (Curies)			2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01	2.45E+02	1.11E+03		5.81E+01
Adjusted Activity (Curies)*			9.29E-01	1.26E+02		6.61E+00	1.80E-01	2.44E+01		1.28E+00	8.12E-03	1.10E+00		5.78E-02	3.67E-04	4.98E-02		2.61E-03
H-3	451	3.26E-04	3.03E-04	4.12E-02	4.15E-02	2.15E-03	5.86E-05	7.97E-03	8.03E-03	4.17E-04	2.65E-06	3.60E-04	3.63E-04	1.88E-05	1.19E-07	1.62E-05	1.64E-05	8.50E-07
C-14	452	1.08E-04	2.52E-02	3.42E+00	3.45E+00	1.79E-01	4.87E-03	6.63E-01	6.68E-01	3.46E-02	2.20E-04	2.99E-02	3.02E-02	1.56E-03	9.93E-06	1.35E-03	1.36E-03	7.06E-05
Sb-125	269	2.01E-07	1.87E-07	2.54E-05	2.56E-05	1.33E-06	3.61E-08	4.91E-06	4.95E-06	2.57E-07	1.63E-09	2.22E-07	2.23E-07	1.16E-08	7.37E-11	1.00E-08	1.01E-08	5.24E-10
Mn-54	473	1.07E-03	9.94E-04	1.35E-01	1.36E-01	7.07E-03	1.92E-04	2.62E-02	2.64E-02	1.37E-03	8.69E-06	1.18E-03	1.19E-03	6.18E-05	3.92E-07	5.34E-05	5.38E-05	2.79E-06
Eu-152	408	1.26E-05	1.17E-05	1.59E-03	1.60E-03	8.33E-05	2.27E-06	3.08E-04	3.10E-04	1.61E-05	1.02E-07	1.39E-05	1.40E-05	7.28E-07	4.62E-09	6.28E-07	6.33E-07	3.29E-08
Fe-55	474	3.47E-01	3.23E-01	8.99E+01	9.03E+01	2.30E+00	6.25E-02	1.74E+01	1.75E+01	4.45E-01	2.82E-03	7.86E-01	7.89E-01	2.01E-02	1.27E-04	3.55E-02	3.56E-02	9.07E-04
Co-60	481	5.72E-01	5.31E-01	2.63E+01	2.68E+01	3.78E+00	1.03E-01	5.09E+00	5.19E+00	7.32E-01	4.64E-03	2.30E-01	2.34E-01	3.30E-02	2.10E-04	1.04E-02	1.06E-02	1.49E-03
Ni-59	None	4.74E-04	4.40E-04	5.99E-02	6.04E-02	3.13E-03	8.52E-05	1.16E-02	1.17E-02	8.06E-04	3.65E-06	5.24E-04	5.28E-04	2.74E-05	1.74E-07	2.37E-05	2.38E-05	1.24E-06
Ni-63	None	7.81E-02	7.26E-02	9.87E+00	9.95E+00	5.16E-01	1.40E-02	1.91E+00	1.92E+00	9.99E-02	6.34E-04	8.63E-02	8.69E-02	4.51E-03	2.86E-05	3.90E-03	3.92E-03	2.04E-04
Nb-94	111	1.64E-06	1.52E-06	2.07E-04	2.09E-04	1.08E-05	2.95E-07	4.01E-05	4.04E-05	2.10E-06	1.33E-08	1.81E-06	1.82E-06	9.47E-08	6.01E-10	8.17E-08	8.23E-08	4.28E-09
Tc-99	141	3.51E-07	3.26E-07	4.44E-05	4.47E-05	2.32E-06	6.31E-08	8.59E-06	8.65E-06	4.49E-07	2.85E-09	3.88E-07	3.91E-07	2.03E-08	1.29E-10	1.75E-08	1.76E-08	9.15E-10

*Adjusted Activity is: 1.15 times measured activity times: 12/70 *0.577 for areas from active core top to top of TS. Factor of 0.577 per Section 6.4. Factor of 12/70 is component to active core length.
Adjusted Activity is: 1.15 times measured activity times: 20/70 * 0.067 for areas from top of TS to 20" above TS. Factor of 0.067 is per Section 6.4. Factor of 20/70 is component length to active core length.
Adjusted Activity is: 1.15 times measured activity times: 45.5/70 * 0.00133 for areas from 20" above TS to top of steam baffle (67" to 112.5").0.000133 factor per Section 6.4. 45.5/70 factor: component to active core length.
Adjusted Activity is: 1.15 times measured activity times: 40/70 * 6.83E-5 for areas above steam baffle (112.5" and above).Factor of 6.83E-5 is per Section 6.4. Factor of 40/70 is component length to active core length.
In addition to the source term factors described above, RV clad has additional divisor of 30 per Design Input 3.13.

Unshielded Dose Rate

	3 Meter Unshielded Dose Rate (mrem/hr)			1 Meter Unshielded Dose Rate (mrem/hr)				
Source Contributor	**3 Meters from RV Side (Core Midplane)**	**3 Meters from RV Bottom (CL)**	**3 Meters from RV Top (CL)**	**1 Meter from Side (Along Core Midplane)**	**1 Meter from RV Top (CL)**	**1 Meter from RV Top (Above Grid Bar Stubs)**	**1 Meter from RV Top (Along RV inner radius)**	**1 Meter from RV Bottom (CL)**
Active Core Region								
Thermal Shield	6.09E+02	5.62E-01	2.46E+00	2.04E+03	8.76E+00	4.30E+00	3.81E+00	1.62E+00
RV Wall/Cladding	2.01E+03	1.00E-05	3.27E-04	8.33E+03	1.21E-03	1.54E-03	1.46E-03	9.34E-05
RV Insulation	1.70E+03	1.28E-09	5.07E-10	7.62E+03	2.04E-07	5.29E-07	5.45E-07	8.97E-08
TS Retainer	5.80E+01	2.08E-04	4.28E-06	1.69E+02	6.96E-05	5.80E-05	4.02E-04	3.57E-03
TS Seal Weights	3.13E+02	8.36E-02	2.97E-01	1.25E+03	1.06E+00	5.30E-01	5.30E-01	2.40E-01
Total Active Core	**4.69E+03**	**6.46E-01**	**2.76E+00**	**1.94E+04**	**9.82E+00**	**4.83E+00**	**4.34E+00**	**1.86E+00**
Above Active Core								
Grid Bar Stubs	1.03E+02	1.56E-02	1.76E+02	5.72E+01	3.24E+02	3.94E+02	3.69E+02	2.60E-02
Thermal Shield	5.02E+01	6.45E-04	8.34E+00	4.16E+01	1.93E+01	1.14E+01	9.41E+00	1.71E-03
TS Seal Weights	1.38E+01	8.53E-05	1.53E-01	1.74E+01	4.56E-01	2.29E-01	2.28E-01	2.34E-04
Top Guide Plate	9.85E+00	1.67E-02	3.85E+01	1.46E+01	6.93E+01	5.44E+01	4.97E+01	2.60E-02
TS Seal Housing	6.55E+00	3.05E-07	4.55E+00	3.11E+00	8.34E+00	9.23E+00	1.01E+01	3.02E-06
Sparger	8.94E-03	5.11E-08	4.13E+00	3.22E-05	7.23E+00	7.85E+00	7.65E+00	7.43E-08
Steam Baffle	5.97E-02	1.05E-06	1.13E+02	2.16E-03	3.54E+02	2.32E+02	2.11E+02	1.64E-06
W/Clad to Top of TS	1.77E+02	3.80E-09	4.92E-03	4.26E-02	1.78E-02		6.32E-03	8.92E-08
Insul to Top of TS	1.57E+02	5.04E-18	1.74E-09	5.20E+02	1.24E-06		5.57E-06	2.97E-14
RV Wall/Cladding to 20" Above TS	3.00E+01	2.82E-09	2.21E-02	4.76E+01	8.84E-02	3.26E-02	2.74E-02	6.53E-08
RV Insulation to 20" Above TS	2.81E+01	7.40E-20	4.89E-09	7.44E+01	3.19E-06		7.89E-06	4.61E-16
W/Clad; 20" Above TS to Baffle	9.53E-01	7.44E-10	3.09E-02	7.56E-01	1.39E-01	4.58E-02	3.58E-02	3.00E-09
Insul; 20" Above TS to Baffle	1.03E+00	1.35E-22	2.78E-07	1.90E+00	1.26E-04	5.41E-05	4.68E-05	1.66E-18
W/Clad; Above Baffle	2.53E-02	1.97E-12	8.12E-02	9.49E-03	3.84E-01	3.84E-01	3.73E-01	5.06E-12
Insul; Above Baffle	2.84E-02	2.85E-30	1.87E-02	4.27E-02	2.77E-02	6.59E-02	8.16E-02	3.80E-21
RV 14" Dia Steam Nozzle	3.15E+00							
Total Above Active Core	**5.81E+02**	**3.30E-02**	**3.45E+02**	**7.79E+02**	**7.83E+02**	**7.10E+02**	**6.58E+02**	**5.39E-02**
Below Active Core								
Thermal Shield (4")	7.62E+00	1.03E-01	2.73E-03	7.00E+00	7.48E-03	4.11E-03	3.68E-03	2.95E-01
Thermal Shield 4 to 10" Below	1.75E+00	8.43E-02	4.72E-04	1.09E+00	1.28E-03	7.19E-04	6.46E-04	1.54E-01
TS Seal Weights	8.28E+00	7.77E-02	7.10E-04	7.21E+00	1.94E-03	9.70E-04	9.70E-04	1.56E-01
TS Retainer	1.16E+01	3.87E-02	2.35E-07	1.28E+01	5.39E-06	4.49E-06	4.49E-06	1.53E-01
Core Support Plate	6.75E-03	5.30E+00	1.43E-06	4.25E-04	2.24E-06	1.79E-06	1.68E-06	2.17E+01
W/Clad, AC to 4" Below AC	5.26E+01	4.78E-05	1.14E-10	1.40E+02	7.35E-09	1.01E-07	1.17E-07	9.59E-05
Insul to 4" Below AC	4.60E+01	8.73E-10	5.65E-19	1.61E+02	9.69E-15	7.78E-12	9.93E-12	1.01E-07
W/Clad, 4" to Bottom of TS R	2.30E+01	5.39E-04	2.24E-11	5.04E+01	1.47E-09	2.37E-08	2.80E-08	1.36E-03
Insul to Bottom of TS R	2.06E+01	1.40E-09	1.09E-19	6.48E+01	1.33E-15	4.75E-13	7.55E-13	1.46E-07
W/Clad to 20" Below TS R	1.17E+00	6.39E-04	8.64E-09	1.73E+00	7.61E-08	9.10E-08	8.85E-08	4.95E-04
RV Insulation to 20" Below TS Retainer	1.11E+00	1.10E-09	9.45E-22	2.83E+00	2.70E-18	1.48E-14	3.81E-14	8.31E-08
W/Clad; 20"to 64" Below TS R	5.03E-02	4.49E-02	5.89E-10	3.63E-02	1.74E-09	1.29E-09	1.20E-09	1.04E-01
Insul; 20" to 64" Below TS R	5.62E-02	1.79E-02	1.03E-23	9.80E-02	1.41E-19	2.31E-15	3.87E-15	6.83E-02
RV 20" Dia Inlet Nozzle	1.44E-01							
Total Below Active Core	**1.74E+02**	**5.67E+00**	**3.91E-03**	**4.49E+02**	**1.07E-02**	**5.81E-03**	**5.30E-03**	**2.26E+01**
TOTAL Unshielded	**5.44E+03**	**6.35E+00**	**3.48E+02**	**2.06E+04**	**7.93E+02**	**7.14E+02**	**6.62E+02**	**2.45E+01**

Shielded Dose Rate at Side — **Side Container Shield Thicknesses: 7"/3"; (7" is along active core region, 16" above & 10" below active core region)**

	Side															
	Contact Dose Rate to RVTS (mrem/hr)				Shielded Dose Rate at 2 Meters from RVTS (mrem/hr)											
Source Contributor	AC Midplane	Just Above Top 7" to 3" Shield Transition	Just Below Bottom 7" to 3" Shield Transition	At Containment 20" Dia Inlet Pen	AC Midplane	37" Below AC Midplane	37" Above AC Midplane	Just Above Top 7" to 3" Shield Transition	Just Below Bottom 7" to 3" Shield Transition	60" Above AC Midplane	74" Above AC Midplane	86" Above Midplane	Steam Baffle Elevation	55" Below AC Midplane	69" Below AC Midplane	81" Below AC Midplane
Active Core Region																
Thermal Shield	9.24E-01	1.25E-02	2.80E-02	2.80E-05	3.29E-01	1.63E-01	2.07E-01	1.20E-01	1.09E-01	8.17E-02	3.61E-02	1.56E-02	3.67E-03	7.35E-02	3.22E-02	1.53E-02
RV Wall/Cladding	4.86E+00	6.18E-02	2.54E-01	1.07E-02	1.70E+00	1.08E+00	1.08E+00	6.84E-01	8.10E-01	5.03E-01	2.65E-01	1.39E-01	3.14E-02	6.12E-01	3.38E-01	1.85E-01
RV Insulation	4.67E+00	4.91E-02	2.15E-01	3.90E-02	1.57E+00	1.02E+00	1.02E+00	6.65E-01	7.80E-01	4.99E-01	2.74E-01	1.50E-01	3.81E-02	6.00E-01	3.44E-01	1.96E-01
TS Retainer	1.29E-03	5.26E-14	1.79E-02	2.46E-05	3.58E-02	5.59E-02	4.35E-03	1.30E-03	4.60E-02	6.35E-04	1.67E-04	4.90E-05	3.58E-06	3.58E-02	1.91E-02	9.44E-03
TS Seal Weights	6.16E-01	8.29E-03	1.87E-02	1.84E-05	1.97E-01	9.76E-02	1.24E-01	7.15E-02	6.53E-02	4.89E-02	2.16E-02	9.35E-03	1.67E-03	4.42E-02	1.94E-02	8.74E-03
Total Active Core	**1.11E+01**	**1.32E-01**	**5.34E-01**	**4.98E-02**	**3.83E+00**	**2.42E+00**	**2.44E+00**	**1.54E+00**	**1.81E+00**	**1.13E+00**	**5.97E-01**	**3.14E-01**	**7.48E-02**	**1.37E+00**	**7.53E-01**	**4.14E-01**
Above Active Core*																
Grid Bar Stubs	3.99E-06	2.33E+01			2.52E-02			1.64E+00		2.11E+00	2.94E+00	2.68E+00	1.11E+00			
Thermal Shield	8.15E-06	1.58E-01	1.31E-15	1.84E-16	1.48E-02	1.07E-03	4.50E-02	4.18E-02	4.33E-04	3.55E-02	1.47E-01	2.32E-01	1.25E-01	2.01E-04	4.84E-05	1.33E-05
TS Seal Weights	6.65E-06	1.57E-02	1.08E-15	1.43E-16	5.40E-03	4.36E-04	1.37E-02	1.17E-02	1.81E-04	9.51E-03	5.40E-03	2.77E-03	2.37E-02	8.54E-05	2.10E-05	5.88E-06
Top Guide Plate	1.26E-05	1.66E-02	1.10E-13	1.56E-14	3.27E-03	3.53E-04	4.70E-03	4.48E-03	1.49E-04	4.49E-03	3.59E-03	5.53E-03	3.28E-02	7.09E-05	1.77E-05	4.92E-06
TS Seal Housing	1.34E-08	1.07E+00	1.41E-18	4.78E-19	1.42E-03	8.09E-05	6.24E-03	6.81E-03	3.15E-05	8.21E-02	1.57E-01	1.07E-01	3.45E-02	1.42E-05	3.33E-06	9.00E-07
Sparger	3.54E-21	8.95E-09			1.00E-05	4.64E-07	1.39E-04	3.31E-04	1.93E-07	4.95E-04	8.85E-04	1.28E-03	1.76E-03	9.45E-08	2.66E-08	8.86E-09
Steam Baffle	1.58E-10	1.09E-04	1.93E-13		1.24E-04	6.68E-06	9.13E-04	1.57E-03	2.62E-06	1.90E-03	2.14E-03	1.80E-03	1.31E-03	1.17E-06	2.71E-07	
W/Clad to Top of TS	6.64E-05	6.61E-01	1.33E-12	1.09E-10	8.68E-02	1.23E-02	2.01E-01	1.90E-01	6.35E-03	1.68E-01	2.07E-01	7.84E-01	8.71E-01	3.63E-03	1.30E-03	5.13E-04
Insul to Top of TS	7.44E-05	7.14E-01	3.02E-09	7.64E-07	8.47E-02	1.39E-02	1.84E-01	1.75E-01	7.57E-03	1.56E-01	1.41E-01	4.99E-01	7.63E-01	4.52E-03	1.74E-03	7.36E-04
RV Wall/Cladding to 20" Above TS	7.99E-08	7.25E+00	1.99E-15	2.50E-13	8.74E-02	9.38E-04	6.47E-01	9.44E-01	4.18E-04	1.04E+00	1.06E+00	8.85E-01	4.79E-01	2.29E-04	7.72E-05	2.94E-05
RV Insulation to 20" Above TS	4.07E-07	7.14E+00	3.15E-11	1.94E-08	1.22E-01	3.20E-03	6.23E-01	8.59E-01	8.01E-04	9.33E-01	9.60E-01	8.38E-01	4.96E-01	3.52E-04	1.13E-04	4.57E-05
W/Clad; 20" Above TS to Steam Baffle	7.78E-13	1.48E-02	2.85E-20		5.87E-03	8.97E-04	2.26E-02	3.31E-02	4.79E-04	3.87E-02	4.68E-02	5.02E-02	4.50E-02			
Insul; 20" Above TS to Steam Baffle	1.50E-10	1.63E-02	9.48E-15	2.43E-11	7.19E-03	1.54E-03	2.30E-02	3.20E-02	9.17E-04	3.67E-02	4.32E-02	4.60E-02	4.15E-02	5.91E-04	2.69E-04	1.34E-04
W/Clad; Above Baffle	2.78E-21	2.47E-08	9.84E-34		3.65E-05	5.30E-06	2.13E-04	4.01E-04	3.10E-06	5.46E-04	8.89E-04	1.26E-03	2.04E-03			
Insul; Above Baffle	7.34E-16	2.49E-07	3.34E-20		5.85E-05	1.08E-05	2.70E-04	4.66E-04	6.74E-06	6.08E-04	9.24E-04	1.25E-03	1.89E-03	4.60E-06	2.34E-06	1.30E-06
RV 14" Dia Steam Nozzle		1.23E+01						8.62E-01		8.62E-01	8.62E-01	8.62E-01	8.62E-01			
Total Above Active Core	**1.73E-04**	**5.27E+01**	**3.05E-09**	**7.84E-07**	**4.44E-01**	**3.47E-02**	**1.77E+00**	**4.80E+00**	**1.73E-02**	**5.48E+00**	**6.58E+00**	**7.00E+00**	**4.89E+00**	**9.70E-03**	**3.59E-03**	**1.48E-03**
Below Active Core																
Thermal Shield 4"	1.76E-06	5.97E-18	1.32E-01	4.73E-05	2.43E-03	6.46E-03	1.74E-04	4.25E-05	6.00E-03	1.88E-05	4.17E-06	1.07E-06	5.95E-08	5.10E-03	1.04E-02	5.94E-02
Thermal Shield 4 to 10" Below	5.38E-08	1.24E-19	3.16E-01	1.03E-04	4.68E-04	1.59E-03	2.81E-05	6.52E-06	1.61E-03	2.83E-06	6.05E-07	1.51E-07	8.15E-09	2.24E-02	3.55E-02	2.31E-02
TS Seal Weights	1.19E-06	3.73E-18	1.16E+00	5.24E-04	2.74E-03	8.36E-03	1.80E-04	4.29E-05	2.09E-02	1.89E-05	4.12E-06	1.05E-06	5.79E-08	6.96E-02	1.11E-01	1.03E-01
TS Retainer	7.32E-07	3.96E-18	1.34E+00	1.13E-03	4.54E-03	9.00E-02	3.23E-04	8.22E-05	1.70E-01	3.76E-05	8.88E-06	2.42E-06	1.57E-07	2.15E-01	2.34E-01	1.92E-01
Core Support Plate	5.83E-12	3.12E-15	1.13E-05	4.70E-04	1.74E-05	1.11E-04	1.19E-06	2.81E-07	1.61E-04					2.05E-04	2.72E-04	2.86E-04
W/Clad, AC to 4" Below AC	4.53E-05	1.44E-13	4.12E-01	8.52E-03	2.91E-02	5.89E-02	4.48E-03	1.67E-03	5.59E-02	9.53E-04	3.37E-04	1.33E-04	1.88E-05	4.96E-02	3.67E-02	1.85E-01
Insul to 4" Below AC	4.61E-05	5.65E-10	2.70E-01	1.97E-02	2.81E-02	5.37E-02	5.00E-03	2.01E-03	5.12E-02	1.19E-03	4.56E-04	1.92E-04	3.11E-05	4.59E-02	3.51E-02	7.35E-02
W/Clad, 4" to Bottom of TS R	2.12E-06	7.25E-15	6.01E+00	2.84E-02	1.00E-02	2.52E-01	1.31E-03	4.66E-04	4.14E-01	2.60E-04	8.97E-05	3.46E-05	4.78E-06	5.06E-01	5.82E-01	5.21E-01
Insul to Bottom of TS R	3.31E-06	6.68E-11	5.70E+00	4.41E-02	9.88E-03	2.44E-01	1.50E-03	5.81E-04	3.73E-01	3.38E-04	1.26E-04	5.20E-05	8.21E-06	4.42E-01	5.05E-01	4.90E-01
W/Clad to 20" Below TS R	1.02E-09	4.47E-18	1.81E-01	6.01E-02	1.08E-02	3.95E-02	9.29E-04	2.40E-04	4.46E-02	9.86E-05	1.09E-05	1.19E-06	8.16E-08	4.68E-02	4.59E-02	4.07E-02
RV Insulation to 20" Below TS Retainer	8.69E-09	2.39E-13	1.84E-01	6.32E-02	1.29E-02	3.69E-02	1.72E-03	6.07E-04	4.09E-02	3.26E-04	9.03E-05	2.15E-05	4.40E-07	4.26E-02	4.19E-02	3.78E-02
W/Clad; > 20" Below TS R	1.28E-14	8.10E-23	1.47E-04	1.10E-02	2.69E-04	1.09E-03	4.71E-05	2.18E-05	1.45E-03	1.43E-05	6.50E-06			1.76E-03	2.29E-03	2.64E-03
Insul; > 20" Below TS R	4.19E-12	8.75E-17	1.89E-04	1.12E-02	3.38E-04	1.13E-03	7.45E-05	3.81E-05	1.45E-03	2.63E-05	1.35E-05	7.44E-06	1.95E-06	1.72E-03	2.16E-03	2.45E-03
RV 20" Dia Inlet Nozzle				1.09E+00					7.90E-02					7.90E-02	7.90E-02	7.90E-02
Total Below Active Core	**1.01E-04**	**6.32E-10**	**1.57E+01**	**1.34E+00**	**1.12E-01**	**7.94E-01**	**1.58E-02**	**5.81E-03**	**1.26E+00**	**3.28E-03**	**1.15E-03**	**4.46E-04**	**6.56E-05**	**1.53E+00**	**1.72E+00**	**1.81E+00**
TOTAL	**1.11E+01**	**5.28E+01**	**1.62E+01**	**1.39E+00**	**4.39E+00**	**3.24E+00**	**4.22E+00**	**6.35E+00**	**3.09E+00**	**6.62E+00**	**7.18E+00**	**7.31E+00**	**4.97E+00**	**2.90E+00**	**2.48E+00**	**2.23E+00**

FINAL

Shielded Dose Rate at Top/Bottom

	Top Dose Rate (mrem/hr)				Bottom Dose Rate (mrem/hr)			
	Contact		2 Meters		Contact		2 Meters	
Source Contributor	**Center Line**	**RV ID***	**Center Line**	**RV ID***	**Center Line**	**RV ID***	**Center Line**	**RV ID***
Active Core Region								
Thermal Shield	2.15E-01	1.49E-01	6.66E-02	4.54E-02	5.16E-02		2.83E-02	
RV Wall/Cladding	6.20E-05	4.36E-05	6.39E-06	4.73E-06	7.59E-06		4.92E-07	
RV Insulation	1.20E-08	8.00E-09	1.42E-10	9.47E-11	4.49E-09		1.47E-10	
TS Retainer	3.99E-06		2.65E-07		1.88E-04		2.11E-05	
TS Seal Weights	3.19E-02	1.60E-02	9.91E-03	4.96E-03	7.68E-03	3.84E-03	4.21E-03	
Total Active Core	**2.47E-01**	**1.65E-01**	**7.65E-02**	**5.04E-02**	**5.95E-02**	**3.84E-03**	**3.25E-02**	
Above Active Core								
Grid Bar Stubs	3.63E+00	8.01E+00	2.57E+00	3.33E+00	5.26E-04		3.22E-04	
Thermal Shield	3.49E-01	3.65E-01	1.99E-01	1.86E-01	7.39E-05		3.10E-05	
TS Seal Weights	1.09E-02	7.32E-03	4.90E-03	3.27E-03	1.03E-05		4.16E-06	
Top Guide Plate	1.22E+00	1.69E+00	8.66E-01	9.57E-01	9.34E-04	2.56E-03	6.62E-04	
TS Seal Housing	1.25E-01	2.21E-01	1.01E-01	1.04E-01	2.10E-07	1.15E-06	2.49E-08	
Sparger	1.42E-02	2.83E-01	1.14E-01	1.52E-01	2.71E-09	2.72E-09	1.93E-09	
Steam Baffle	1.06E+01	5.39E+00	4.73E+00	3.16E+00	6.61E-08	1.58E-07	4.16E-08	
W/Clad to Top of TS Seal	4.35E-04		1.57E-04		8.58E-09		4.26E-10	
Insul to Top of TS Seal	1.82E-07	1.21E-07	5.40E-10	3.60E-10	1.94E-14		4.83E-18	
RV Wall/Cladding to 20" Above TS	1.88E-03	1.38E-03	7.95E-04	5.45E-04	4.42E-09	2.95E-09	4.58E-10	3.05E-10
RV Insulation to 20" Above TS	3.33E-07	2.22E-07	1.45E-09	9.67E-10	3.21E-16		7.23E-20	
W/Clad; 20" Above TS to Baffle	6.93E-04	1.02E-03	1.25E-03	8.82E-04	1.74E-10		4.07E-11	
Insul; 20" Above TS to Baffle	1.02E-06	6.82E-07	6.88E-08	4.59E-08	1.63E-18		1.84E-22	
W/Clad; Above Baffle	4.37E-05	4.35E-03	3.56E-03	3.85E-03	2.27E-13		9.23E-14	
Insul; Above Baffle	6.72E-08	8.79E-06	6.95E-04	1.80E-03	1.65E-21		1.64E-30	
RV 14" Dia Steam Nozzle								
Total Above Active Core	**1.60E+01**	**1.60E+01**	**8.59E+00**	**7.90E+00**	**1.54E-03**	**2.56E-03**	**1.02E-03**	**3.05E-10**
Below Active Core								
Thermal Shield 4"	2.14E-04	1.44E-04	8.08E-05	5.43E-05	7.74E-03	6.67E-03	5.22E-03	4.09E-03
Thermal Shield 4 to 10" Below	3.66E-05	2.46E-05	1.39E-05	9.33E-06	3.12E-03	1.61E-02	3.56E-03	8.17E-03
TS Seal Weights	5.55E-05		2.10E-05		3.40E-03	2.90E-02	3.40E-03	9.61E-03
TS Retainer	3.73E-07	3.11E-07	1.70E-08	1.42E-08	1.21E-03	5.28E-03	1.66E-03	2.48E-03
Core Support Plate	5.47E-08	3.63E-08	3.40E-08	2.75E-08	8.50E-01	8.92E-04	3.85E-01	1.68E-01
W/Clad, AC to 4" Below AC	7.54E-10	5.03E-10	1.26E-11	8.40E-12	1.68E-06	5.08E-06	2.39E-06	2.48E-06
Insul, AC to 4" Below AC	5.68E-15	1.36E-12	1.03E-18	1.48E-17	5.80E-09	3.87E-09	7.91E-11	5.27E-11
W/Clad, 4" to Bottom of TS R	1.52E-10	1.01E-10	2.53E-12	1.69E-12	2.42E-05	3.87E-05	2.44E-05	1.96E-05
Insul to Bottom of TS R	6.90E-16		1.56E-19		1.15E-08	7.67E-09	1.32E-10	8.80E-11
W/Clad to 20" Below TS R	3.08E-09	2.05E-09	4.62E-10	3.08E-10	5.08E-06	1.15E-05	1.36E-05	1.00E-05
RV Insulation to 20" Below TS Retainer	3.46E-19		6.13E-22		2.30E-09	1.54E-09	3.84E-10	2.56E-10
W/Clad; > 20" Below TS R	5.12E-11	3.43E-11	1.76E-11	1.17E-11	1.09E-07	4.04E-03	2.99E-03	4.06E-03
Insul; > 20" Below TS R	1.20E-19		1.13E-23		7.92E-11		1.26E-03	
RV 20" Dia Inlet Nozzle						1.09E+00		7.90E-02
Total Below Active Core	**3.07E-04**	**1.69E-04**	**1.16E-04**	**6.37E-05**	**8.66E-01**	**1.15E+00**	**4.03E-01**	**2.75E-01**
TOTAL	**1.62E+01**	**1.61E+01**	**8.67E+00**	**7.95E+00**	**9.27E-01**	**1.16E+00**	**4.37E-01**	**2.75E-01**

* The dose rate at the reactor vessel inside diameter is the sum of the QAD dose rate result and 2/3 the corresponding centerline dose rate value.
The QAD model considers 1/3 the annular source. The factor of 2/3 is the dose contribution attributed to the annular source not included in the QAD model.

BNFL Inc.

Big Rock Point Restoration Project
Major Component Removal
10269 US 31 North
Charlevoix, MI 49720-9436
Tel: (231) 547-8357
Fax: (231) 547-8342

Mr. B Slimp
Sargent & Lundy
55 East Monroe Street
Chicago, IL
60603-5780

Your ref:
Our ref: BRP-2000-05-228
WBS: 1.3.04

May 25, 2000

Dear Mr. Slimp:

Big Rock Point Restoration Project - 5339 – Reactor Vessel Grid Bars and Neutron Windows

I write to confirm that the Reactor Vessel (RV) top guide grid bars and neutron windows will be removed from the RV prior to packaging and shipment of the RV.

Please ensure all S&L activities associated with the Reactor Vessel Transportation System (RVTS) are cognizant of this fact.

Yours sincerely,

MD Papp
Engineering Manager
Reactor Decommissioning Group

c: D Jew. file

CALCULATION N-10525-020-0002

REVISION 1

ATTACHMENT "M" NOT INCLUDED

APPENDIX 5-2

BIG ROCK POINT

REACTOR VESSEL AND INTERNALS

CHARACTERIZATION AND CLASSIFICATION

Report WMG-9902 Rev. 1

June 1999

WMG Project 8057

Prepared for:

Consumers Energy

Prepared by:

WMG, Inc.
16 Bank Street
Peekskill, NY 10566

FOREWORD

This report summarizes the work performed by WMG, Inc. to support the decommissioning of Big Rock Point for Consumers Energy. This work was performed under Contract C0029076.

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1.0 INTRODUCTION AND SUMMARY

In September 1998, WMG was engaged by Consumers Energy to support the Big Rock Point Decommissioning Project. WMG's task was to characterize and classify the Big Rock Point reactor vessel and internals components. The results of WMG's analytical work are summarized below. These results are based on analysis as well as empirical data in the form of radiation level measurements of vessel internals. The results provide a regulatory basis for proceeding with disposition of the reactor vessel and internals.

The neutron transport and activation computer based analytical methods used for this Project are discussed in section 2.0. The ORIGEN2[1] results from these calculations were normalized to radiation level measurements taken on internals components. The normalized results are presented in section 3.0. The component specific results in terms of estimated activity and 10 CFR Part 61 status as of September 1, 2002 are presented in section 4.0. Estimated errors associated with the source term calculations and normalization are discussed in section 5.0. References are presented in section 6.0

The methodology used one-dimensional neutron transport calculations using ANISN[2]. The ANISN results were used as inputs to the ORIGEN2 activation code. The analytical results were then normalized to measured radiation survey results using detailed point kernel shielding models with QAD-CGGP[3]. The component specific results in terms of estimated activity including surface contaminants and 10 CFR Part 61 status as of September 1, 2002 are summarized in Table 1-1 below.

Table 1-1
Component Summary September 2002

Component	Total Curies	Co-60 Curies	10 CFR 61 Table 1	10 CFR 61 Table 2
Steam Baffle	1.09E+01	4.95E+00	0.02	< 0.01
Core Spray Sparger	8.78E-01	3.97E-01	0.03	< 0.01
Seal Housing	1.45E+02	6.60E+01	0.04	0.05
Top Guide Plate	1.17E+03	5.38E+02	0.41	0.54
Thermal Shield	7.45E+03	3.83E+03	0.18	0.20
Thermal Shield Ret.	1.18E+02	5.94E+01	0.07	0.08
Seal Weights	8.54E+02	5.89E+02	0.08	0.04
Core Support Plate	1.40E-01	1.36E-1	0.01	< 0.01
Inlet Diffuser	2.24E-2	2.18E-2	0.02	< 0.01
Inlet Baffle	1.63E-01	1.59E-01	< 0.01	< 0.01
Cladding	2.09E+02	1.30E+02	0.04	0.01
Wall	5.90E+02	1.52E+02	< 0.01	< 0.01
Insulation	5.21E+01	3.05E+01	< 0.01	< 0.01
Neutron Windows	1.37E+04	6.92E+03	2.09	2.67
Subtotal Vessel Intact	2.43E+04	1.23E+04		
GTCC Material				
Top Guide	4.84E+04	2.12E+04	8.56	11.87

These results are separated into those components which can be shipped intact within the reactor vessel, and components which are Greater Than Class C (GTCC) material. As shown in Table 1 above, the only component considered to be GTCC is the top guide, excluding the peripheral top guide plate. As of September 1, 2002 this component, weighing about 1,530 lbs. will have an activity of about 48,400 curies and is more than ten times the Class C limits for disposal. The top guide will have to be removed, segmented and packaged for on-site storage as GTCC materials.

The remaining components can be disposed as low-level radioactive waste (LLRW) consistent with the USNRC Branch Technical Position (BTP). The vessel plus the internals will satisfy the current Barnwell disposal limit of 40,000 curies. The analytical results indicated that the reactor vessel plus the remainder of the internals would contain about 24,300 curies as of September 1, 2002. Most of this activity, about 13,700 curies, is in four neutron windows. For intact vessel shipment, the four neutron windows may have to be relocated to the center of the vessel to reduce the dose rates on the vessel exterior.

The results also indicate that the stainless steel based insulation around the vessel exterior contains about 50 curies. Although detailed calculations of external dose rates were not within the scope of this Project, it is expected that the dose rate contribution from the insulation will prevent compliance with the DOT requirement of less than 1 R/hr at 3 meters. The vessel itself, without insulation, may exceed 1 R/hr at 3 meters, which would preclude licensing as an IP2 package under a DOT exemption. We recommend that further analytical work, coupled with best available empirical data, be used to validate the dose rate at 3 meters from the unshielded vessel. The worst case scenario would require package licensing under 10 CFR Part 71 or vessel insulation removal before shipment under a DOT exemption.

2.0 NEUTRON TRANSPORT AND ACTIVATION

The ANISN computer program was used to estimate neutron fluxes at various radial and axial locations in the vessel using the SAILOR[4] cross-section library. The resultant fluxes were then used as inputs to the ORIGEN2 computer program to perform activation analysis on individual components. This section defines the parameters used for input to these computer programs, discusses the models developed for the project and summarizes program output.

The same ANISN/ORIGEN2 methodology has been used to support decommissioning activities at Shoreham, Yankee Rowe, Maine Yankee, Trojan and Saxton. The methodology has been benchmarked relative to actual component-specific radiation surveys, and found to provide reasonable characterization results for the components of interest. This methodology has been refined to minimize the conservatism found during earlier projects and represents the best available information for project planning.

For this project, the ORIGEN2 results were normalized to radiation level measurements taken on internals components as discussed in section 3.0. Benchmarking to empirical data reduces the uncertainties and conservatism in the activation analysis results.

2.1 Input Parameters

Use of the ANISN and ORIGEN2 computer programs required that the components located within the pressure vessel be accurately modeled in terms of their physical characteristics and exposure histories. These input parameters are summarized below.

2.1.1 Component Physical Parameters

Table 2-1 identifies the components used to model the pressure vessel, internals and the referenced basis for their characteristics. The information and referenced drawings in Table 2-1 were used to determine the inputs to model the reactor vessel and internals in the ANISN program axial and radial models. Figure 2-1 shows a cutaway sketch of the Big Rock Point reactor pressure vessel and internals

2.1.2 Material Compositions

Elemental material compositions are required inputs to the ANISN, ORIGEN2, and QAD-CGGP programs. All the reactor vessel internal components of interest are fabricated from type 304 stainless steel. The reactor pressure vessel itself is fabricated of carbon steel with type 304 stainless steel cladding. Initial material composition data was taken from NUREG CR 3474[5]. The 304 stainless steel and carbon steel data is summarized in Table 2-2 for the precursor elements of interest.

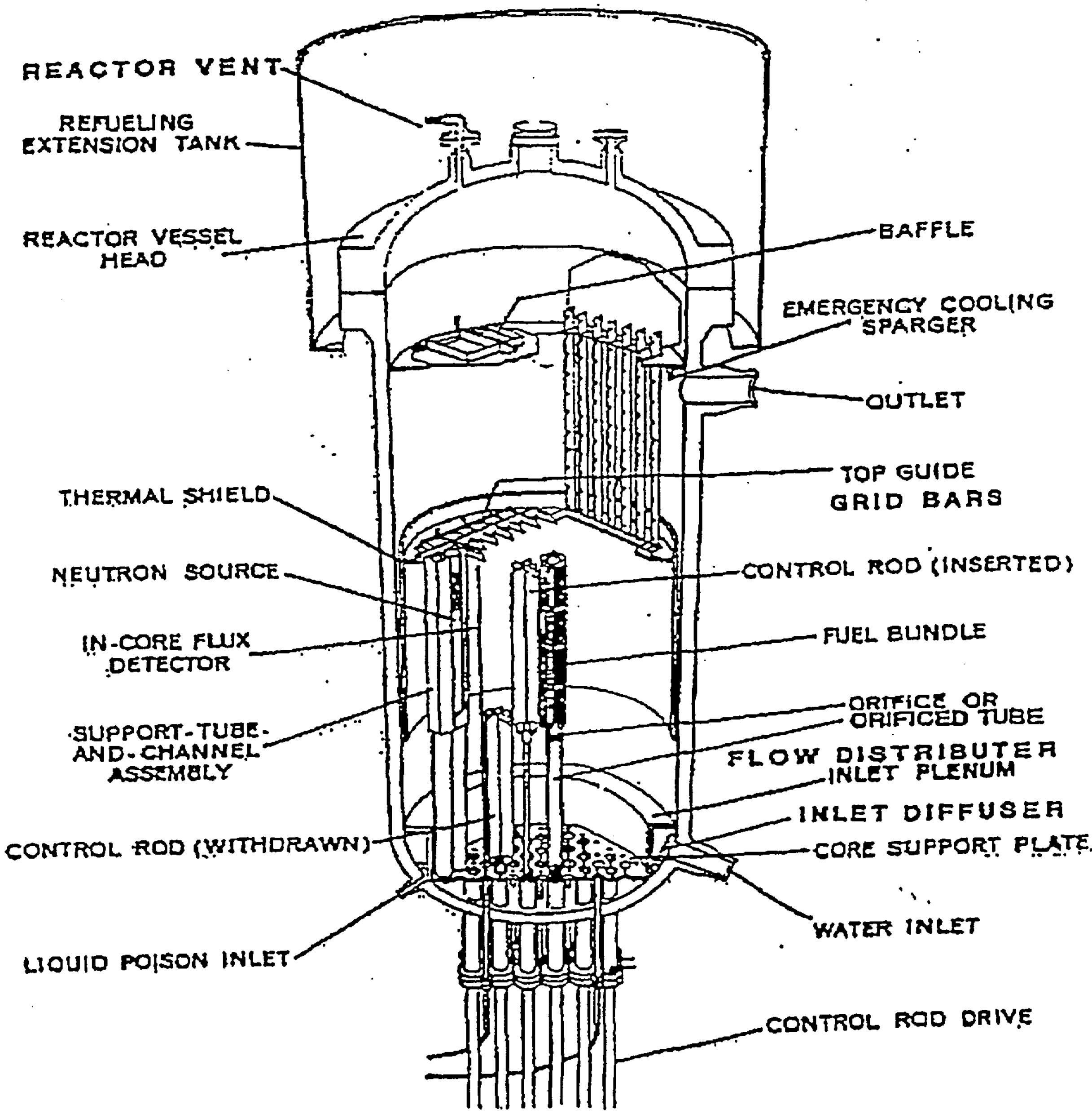
TABLE 2-1

Big Rock Point Reactor Pressure Vessel and Internals

	Drawing Numbers	Length (inches)	Width / Thickness (inches)	Height (inches)	OD (inches)	Volume (ft³)	Weight (lbs)
GTCC Material							
Top Guide	197E861					3.05E+00	1.53E+03
					Subtotal	**3.05E+00**	**1.53E+03**
LLRW Internals							
Steam Baffle	794E830		1.000		105.500	3.27E+00	1.64E+03
Sparger	M248 Sh46-1,41-2		2" SCH 40 PIPE		105.000	2.60E-01	1.30E+02
Top Guide Plate	197E118		1.000		99.188	1.63E+00	8.13E+02
Seal Housing	706E276			3.000	105.690	2.15E+00	1.07E+03
Thermal Shield	197E853		1.500	92.000	103.000	2.59E+01	1.30E+04
Thermal Shield Retainer (6 units)	237E956	10.000	0.750	14.130		1.09E+00	5.47E+02
Seal Weights (12 units)	706E276, 104R175	18.000	1.000	27.000		1.02E+01	5.08E+03
Neutron Windows (4 units)	107C353-9		0.718	72.000	6.450	2.46E+00	1.23E+03
Core Support Plate	762-D229, E201-809-3		1.500		79.250	5.08E+00	2.54E+03
Inlet Diffuser (2 units)	795E369	27.000	5.500	0.375		3.97E-01	1.98E+02
Inlet Baffle	795E421			36.000		4.64E+00	2.32E+03
					Subtotal	**5.71E+01**	**2.85E+04**
Reactor Vessel Assembly							
Reactor Vessel Assembly	197E853, 104R175, F 230-791-2			287.625	116.938	4.14E+02	2.03E+05
Reactor Vessel Insulation	795E369			290.625	122.500	2.59E+02	5.18E+03
					Subtotal	**6.73E+02**	**2.08E+05**

Other material types present in the vessel internals were also incorporated into the neutron transport models. These additional materials included zircaloy-2, inconel, ordinary concrete and vessel insulation of type 304 stainless steel at 20 lbs/ft³.

FIGURE 2-1
Big Rock Point Reactor Pressure Vessel and Internals



REACTOR VESSEL & INTERNALS

TABLE 2-2

Initial Material Composition Data

Elements	**304 SST Weight %**	**Carbon Steel Weight %**
NITROGEN	0.045	0.008
CHROMIUM	18.400	0.000
MANGANESE	1.530	1.350
IRON	70.600	97.570
NICKEL	10.000	0.610
MOLYBDENUM	0.260	0.580
NIOBIUM	0.009	0.002
COBALT	0.141	0.012

Data taken from
NUREG CR-3474

2.1.3 Plant Operating History

A detailed operating history for the 29 cycles experienced at Big Rock Point was used as input to the ORIGEN2 neutron activation program. The operating history is summarized in Table 2-3.

TABLE 2-3
Big Rock Point Operating History

Cycle	Start (Date)	Stop (Date)	Length (Days)	Outage (Days)	Total (Days)	Net MW-hrs E
Critical	09/27/62					
A	02/13/63	12/01/63	291	20	311	
B	12/21/63	09/18/64	272	0	272	
C	09/18/64	09/04/65	351	0	351	314,949
1	09/04/65	04/09/66	217	29	246	318,501
2	05/08/66	09/17/66	132	53	185	143,982
3	11/09/66	05/19/67	191	26	217	247,464
4	06/14/67	02/11/68	242	33	275	369,782
5	03/15/68	06/21/68	98	25	123	118,020
6	07/16/68	04/18/69	276	21	297	383,919
7	05/09/69	02/13/70	280	44	324	320,895
8	03/29/70	02/12/71	320	29	349	358,818
9	03/13/71	03/18/72	371	56	427	385,848
10	05/13/72	03/02/73	293	45	338	362,983
11	04/16/73	03/23/74	341	43	384	437,737
12	05/05/74	06/02/74	28	55	83	35,159
13	07/27/74	01/31/76	553	179	732	539,025
14	07/28/76	07/23/77	360	93	453	486,908
15	10/24/77	02/02/79	466	276	742	526,580
16	11/05/79	10/31/80	361	96	457	479,711
17	02/04/81	02/05/82	366	68	434	502,456
18	04/14/82	05/13/83	394	104	498	503,238
19	08/25/83	05/31/84	280	55	335	359,977
20	07/25/84	09/06/85	408	63	471	534,304
21	11/08/85	01/02/87	420	66	486	565,651
22	03/09/87	04/08/88	396	80	476	496,899
23	06/27/88	06/09/89	347	65	412	480,297
24	08/13/89	09/21/90	404	68	472	576,735
25	11/28/90	11/29/91	366	169	535	537,257
26	05/16/92	06/26/93	406	71	477	513,737
27	09/05/93	09/30/94	390	59	449	549,032
28	11/28/94	01/05/96	403	92	495	565,088
29	04/06/96	08/30/97	511	472	983	556,924

2.2 ANISN Discrete Ordinates Neutron Transport Calculations

The ANISN code was used to develop one-dimensional radial and axial neutron transport models. The ANISN cylindrical source geometry was used for the radial models, and the slab geometry was used for the axial models. Both radial and axial models were forward solutions using a P3 maximum order of scatter and an S8 order of angular quadrature.

2.2.1 ANISN Radial Models

The ANISN radial models consisted of the reactor core, neutron windows, thermal shield, seal weights, reactor vessel cladding, vessel wall, vessel insulation, cooling jacket tank, and biological shield. All air and water gaps were included in the model. The one-dimensional radial model utilized a radially-averaged, flat core flux of 2.49E+14 n/cm^2-s determined from ORIGEN2 fuel burn-up calculations.

The variations in the water densities in the axial direction were addressed in ten regions. Core inlet water density was used for the lower internals regions. The active fuel section was broken into nine axial regions based on water quality and void fraction information provided by Consumers Energy. The upper region of the reactor used a water/steam mixture corresponding to the nominal reactor exit quality.

The reactor core section of the radial ANISN model was broken into three regions based on an out-in fuel management strategy. This fuel loading results in a relatively uniform radial power distribution. Unlike most plants, a low leakage refueling strategy was not employed at Big Rock Point. Therefore, modeling the core with the fresh fuel on the periphery is appropriate. Big Rock Point also employed a fuel management strategy where some fuel batches were burned for three cycles and other fuel batches were burned for four cycles. For the neutron transport analysis WMG used the three cycle fuel batches. The three cycle fuel batch increases the amount of fresh fuel on the periphery versus the four batch loading and therefore yields more conservative results.

The structural components within the core region over the 70 inch active fuel height including fuel cladding, spacer grids, guide tubes, etc. were obtained from DOE/RW-0184[6] and discussions with Consumers Energy personnel.

Three radial model ANISN calculations were performed. All three models assumed a radially-averaged, flat core power distribution, with the total

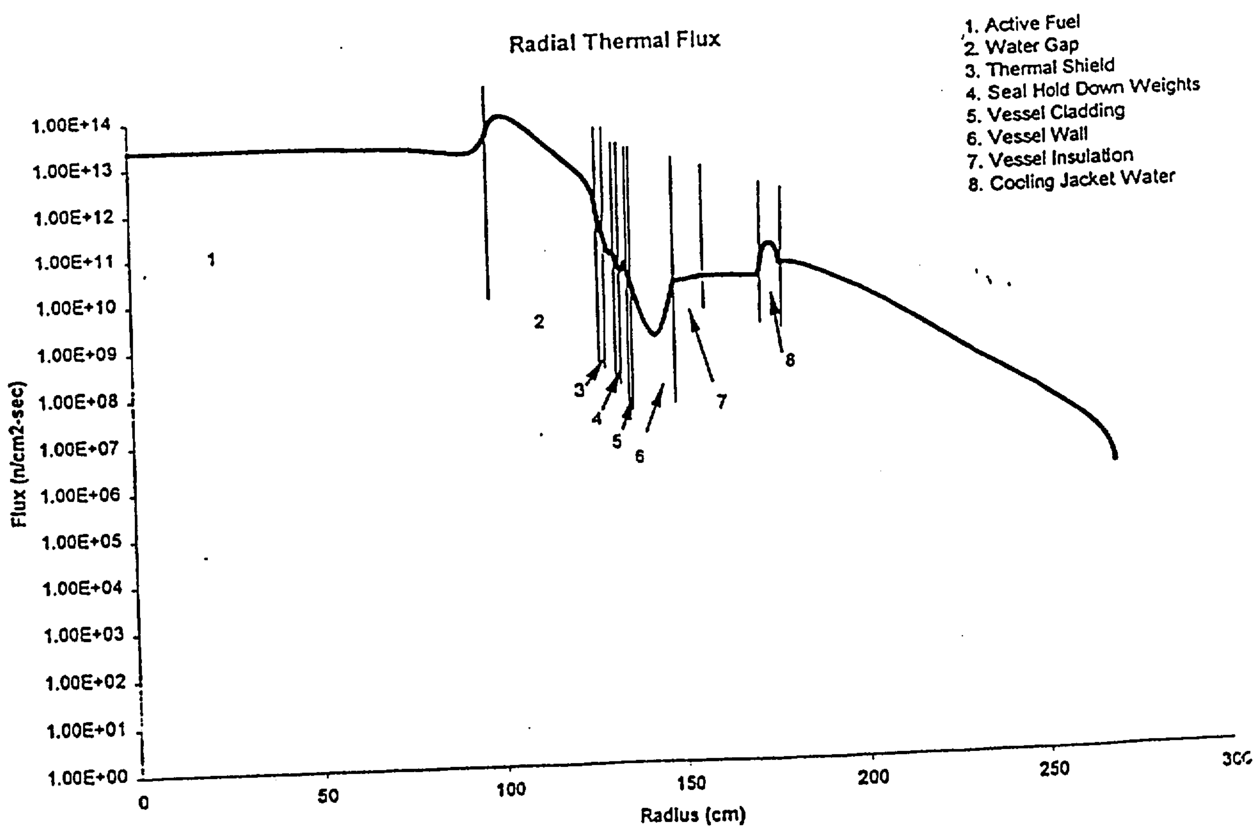
neutron source equal to 2.49E+14 n/cm^2-s, calculated by ORIGEN2 for an average Big Rock Point core.

The first radial model included the seal weights, which cover the majority of the thermal shield periphery.

The second radial model excluded the seal weights present in the first model to determine their effects on the flux in the surrounding components. A weighted average of the results from the first and second models was used to accurately determine the average flux in each radial component. The radial thermal flux results are shown in Figure 2-2.

The third radial model included neutron windows. This model was to determine the flux characteristics in the neutron windows and the impact on the flux levels in the components radially behind the neutron windows.

FIGURE 2-2
Big Rock Point Radial Thermal Flux Distribution



2.2.2 ANISN Axial Models

Two axial models were run for the project. Because of array size restrictions with the ANISN program, in order to provide detailed axial component models, separate axial component models were run for the components above and below the active core region. The first model was an axial component model to determine the flux levels and energy spectra at the axial locations of interest below the active core with a water/steam reflector above the core. The second axial component model determined flux levels and energy spectra above the core and utilized a water reflector below the active fuel.

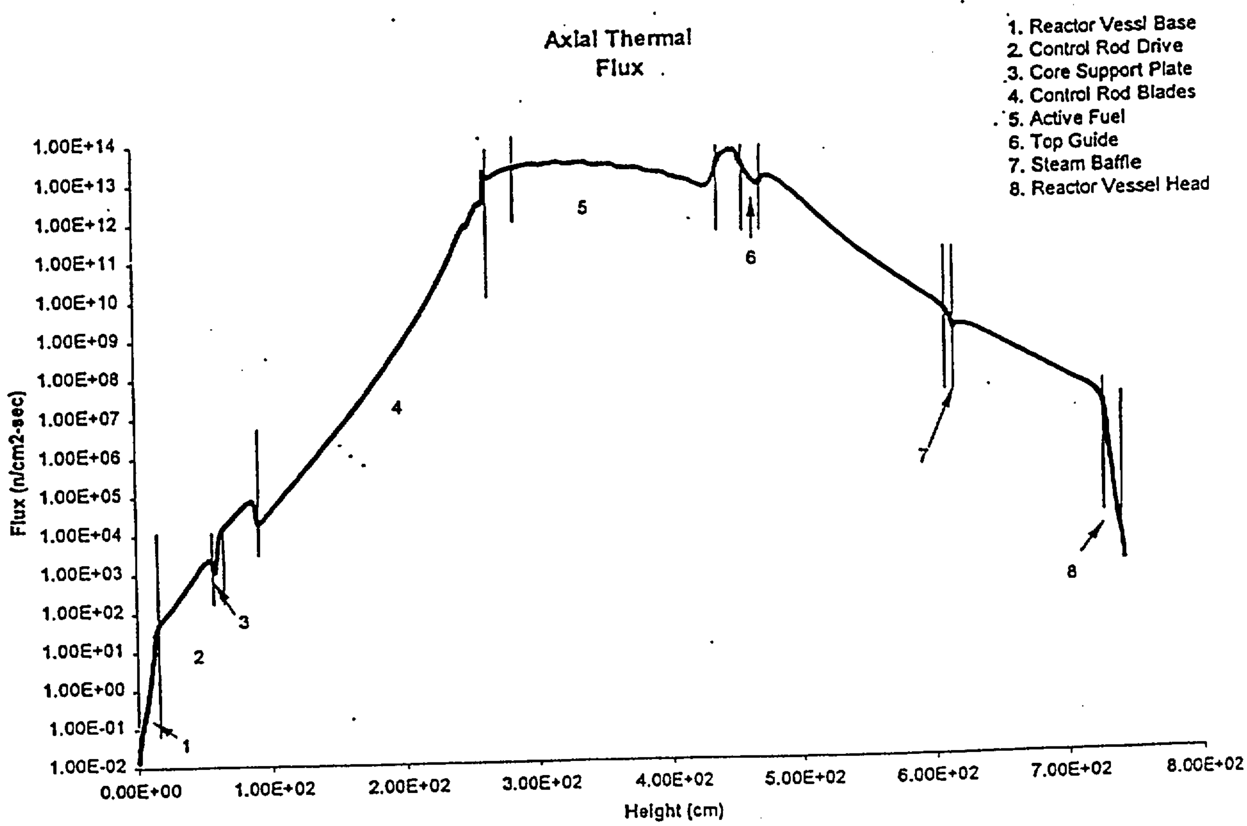
The ANISN axial component models were more complex than the radial models because of the number of components present and the significant variations in moderator densities, and water to metal volume fractions in the axial directions. Ten different moderator densities were used and heterogeneous structural components were modeled as homogeneous regions containing the appropriate stainless steel and water fractions.

The volume averaged fuel parameters of the three radial core regions were used as a baseline for the core fuel regions in the axial models. The active fuel was then broken into nine axial nodes based on thermal hydraulic variations in void fraction and water density.

After a detailed review of component physical characteristics, 28 distinct axial zones were identified and incorporated into the axial models. The axial models only included components within the core equivalent radius above and below the core. The 32 CRBs were modeled in their fully withdrawn position for the lower component model. The thermal flux results as a function of axial height are plotted in Figure 2-3.

Figure 2-3

Big Rock Point Axial Thermal Flux Distribution



2.3 ORIGEN2 Point Neutron Activation Calculations

The ORIGEN2 computer code was used to calculate the activation and depletion of radionuclides in components exposed to neutrons. Neutron cross-sections from the BWRUS[7] library were used for the core activation, and neutron cross-sections from the thermal cross-section library were used for the thermal activation.

Based upon the ANISN spectral results, the core and thermal ORIGEN2 input fluxes were weighted to obtain a location-specific neutron spectrum for each component or region of interest. The thermal values were adjusted for the local area temperatures to reflect the reduced activation cross-sections at elevated temperatures.

3.0 NORMALIZATION OF ORIGEN2 ACTIVATION CALCULATIONS

Detailed radiation profiles of the internals were used to normalize the ORIGEN2 calculated radionuclide concentrations. The radiation levels were used in conjunction with QAD-CGGP and MegaShield™ models to determine Co-60 content at the time of radiation measurement. Scaling factors from ORIGEN2 were then used to determine the remaining curie content of the component. WMG uses the same methodology to characterize control rod blades, reactor control assemblies, shroud head bolts and other various types of reactor hardware components that are disposed of on a regular basis. Therefore, applying the same methodology to a reactor pressure vessel and internals for disposal characterization and classification is appropriate.

This section describes the radiation measurements obtained and how they were employed to normalize the ORIGEN2 calculated Co-60 concentrations.

3.1 Radiation Survey Plan

Part of the scope of work for the reactor pressure vessel (RPV) and internals characterization entailed obtaining detailed radiation profiles on the reactor vessel internal components. These survey results were used to benchmark and normalize the activation analysis results obtained from ANISN/ORIGEN2 calculations. The normalization to empirical data significantly reduced the uncertainties associated with the activation analysis calculations. The survey results were used in conjunction with a series of detailed QAD-CGGP point kernel shielding calculations to normalize the activation results. The survey plan and surveys results are included in Appendix A.

The surveys were performed using a Dositec PR-2 with a 12 inch circular standoff. Because the Dositec PR-2 was calibrated using a Cs-137 source, all measured dose rates were corrected by a factor of 1.25. This is done in accordance with the Palisades Nuclear Plant Health Physics Procedure[8] to correct for the higher energy Co-60 gammas, assumed to be the only dose contributor. Separate surveys listed below provided the empirical data used for the analysis normalization.

- Axial distribution of activity
- Azimuthal distribution of activity
- Radial distribution of activity (top guide)
- Activation of the neutron windows

3.1.1 Axial Distribution Surveys

Radiation profiles were obtained along the inner surface of the RPV and internals at the south side of the reactor pressure vessel (away from the open top guide) at the south-southeast and south-southwest locations as shown in Figure 3-1 (DWG # 104R175 Sh. 3). Radiation levels were obtained using the Dositec PR-2 with the standoff at 1 foot axial elevation increments and are shown in Figure 3-2. The peak at the 609 foot survey location is because the probe was on contact with the top guide plate.

Figure 3-1
Axial Distribution Survey Location

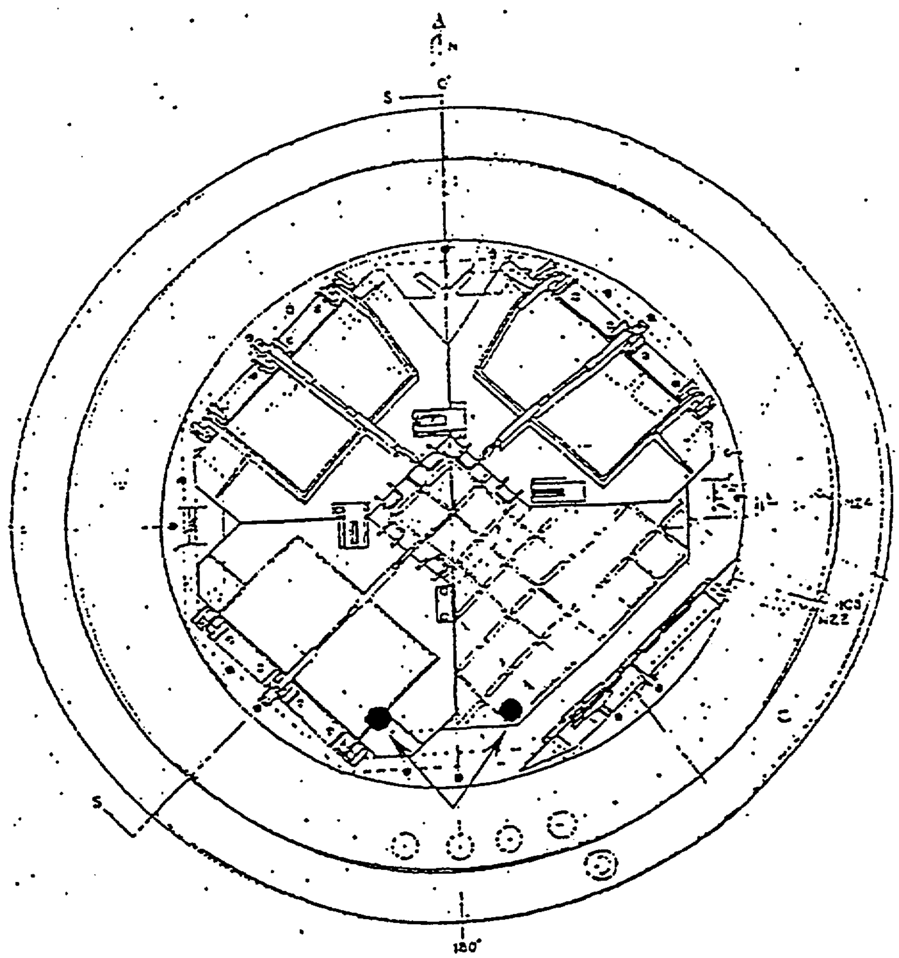
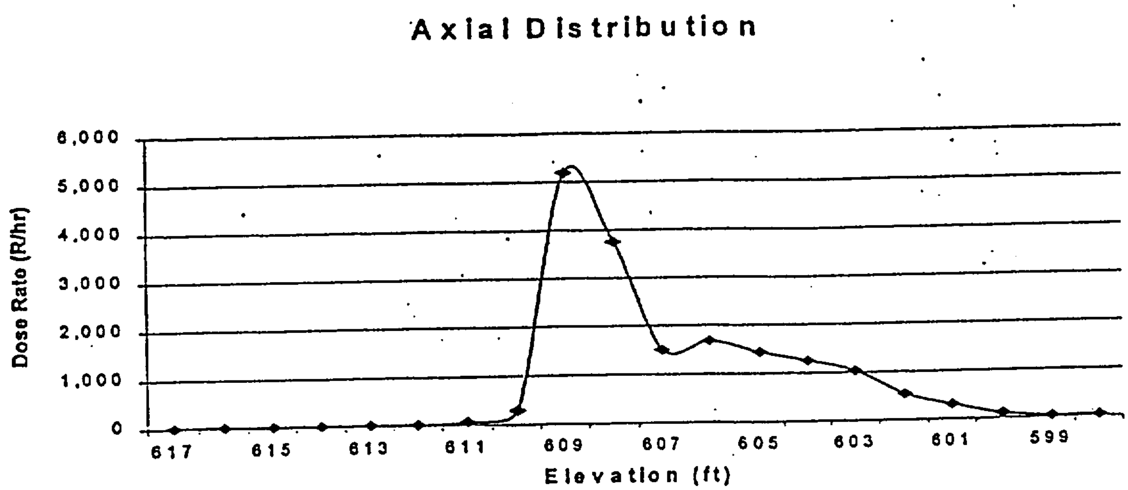


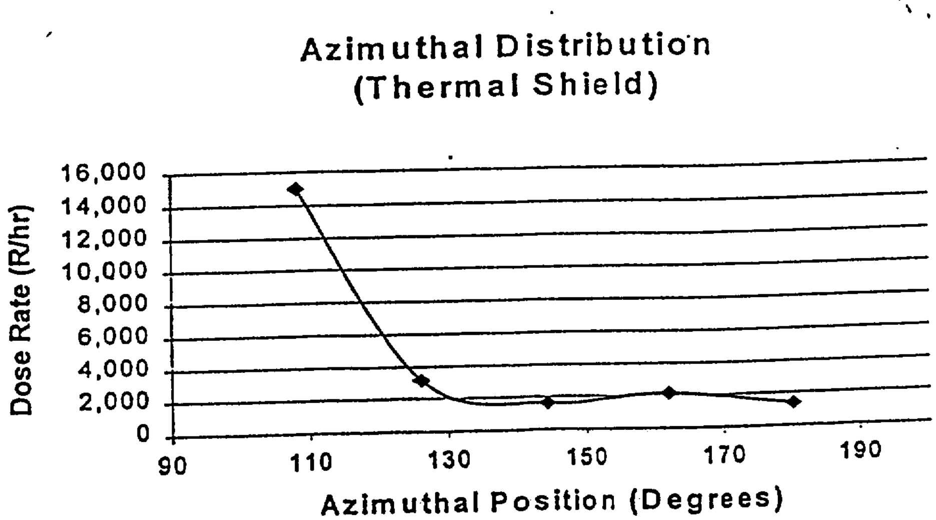
Figure 3-2



3.1.2 Azimuthal Distribution Surveys

Azimuthal surveys were taken every 18 degrees from 108 to 180 degrees at the 604 foot and 606 foot axial locations. The 90 degree survey result was taken from the neutron window survey. A J-hook was required to hold the standoff in contact with the thermal shield. The azimuthal activity is greatly skewed toward the east (90 degree) location for two reasons. First, the thermal shield is more activated behind the neutron windows, and second, a large contribution of the total dose can be attributed to the neutron window activity. The azimuthal results are shown in Figure 3-3

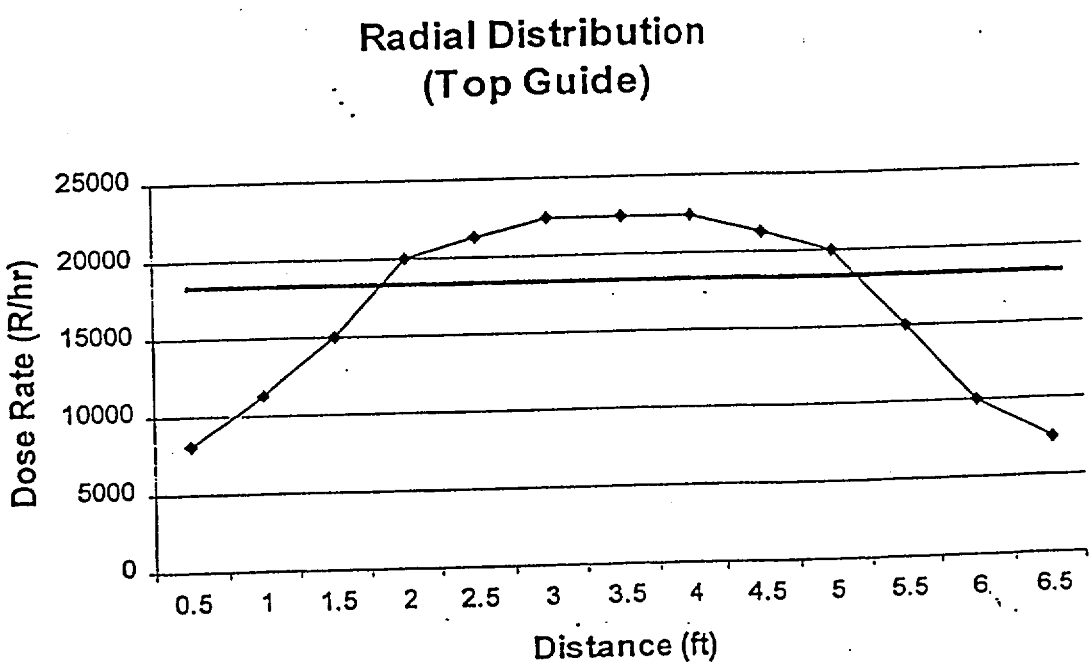
Figure 3-3



3.1.3 Radial Distribution Surveys (Top Guide)

The dose rates for the top guide are the most representative of the radial distribution of the neutron flux within the Big Rock Point core, because of their horizontal positioning during plant operation. However, at the time the dose measurements were taken, the top guide was in its open (vertical) position. Dose rate measurements started at the bottom center of the top guide and were taken in 6 inch axial increments along the length of top guide bars. The standoff was in contact with the center and its adjacent guide bar for all measurements. The radial distribution survey results are shown in Figure 3-4.

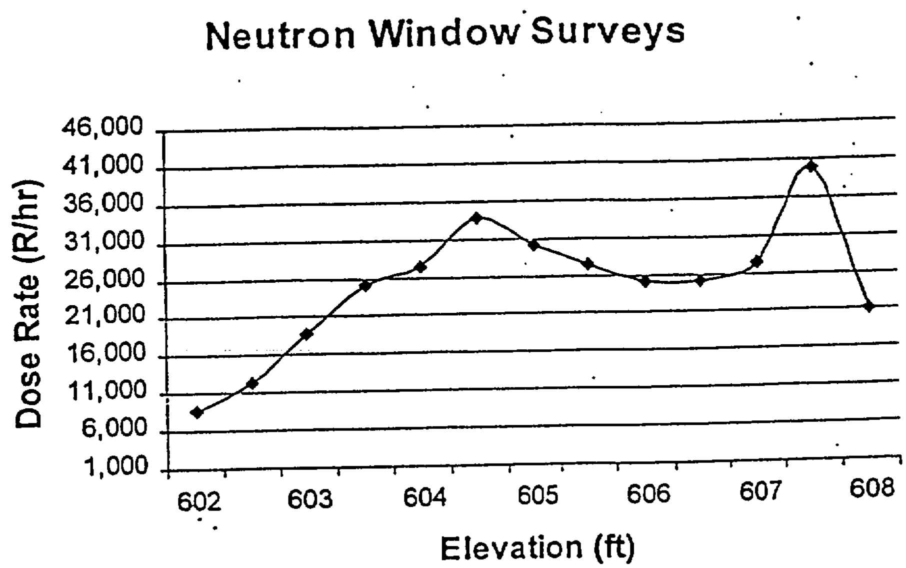
Figure 3-4



3.1.4 Neutron Window Surveys

Surveys started at the bottom of the neutron windows and were taken every 6 inches axially. The standoff was in contact with both neutron windows for all but one measurement, where a bracket slightly offset the detector location. Peaks in the window surveys occur as the probe passes each of the stainless steel brackets that mount the neutron windows to the thermal shield. The neutron window survey results are shown in Figure 3-5.

Figure 3-5



3.2 Normalization Methodology

The ORIGEN2 activity concentration was normalized to vessel internal survey results performed by WMG and Consumers Energy. The activation results were normalized via complex point kernel calculations using QAD-CGGP and MegaShield™.

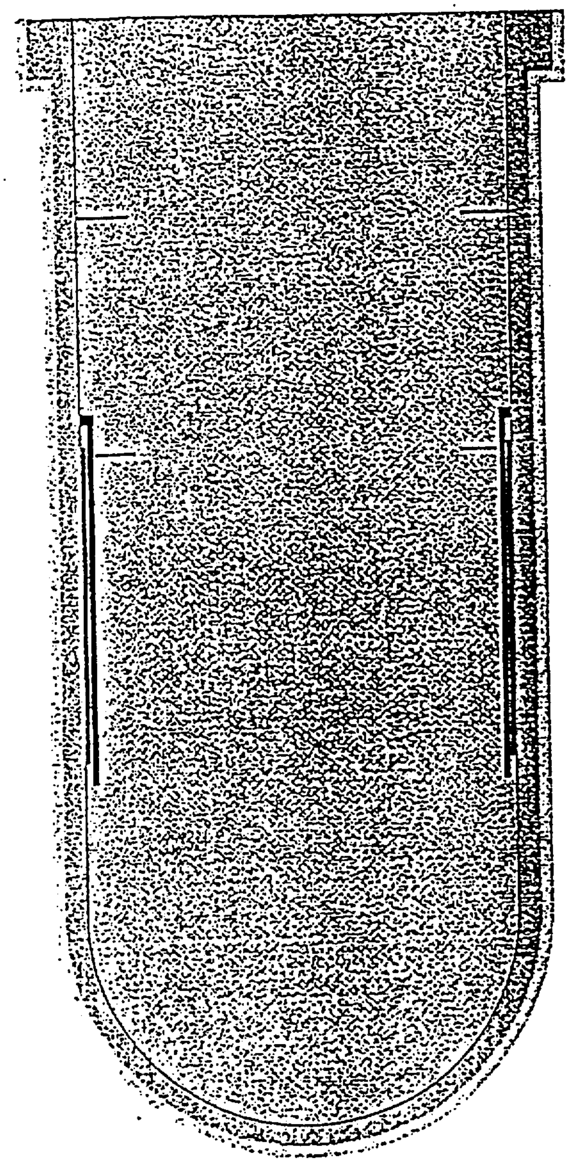
3.2.1 QAD-CGGP Models

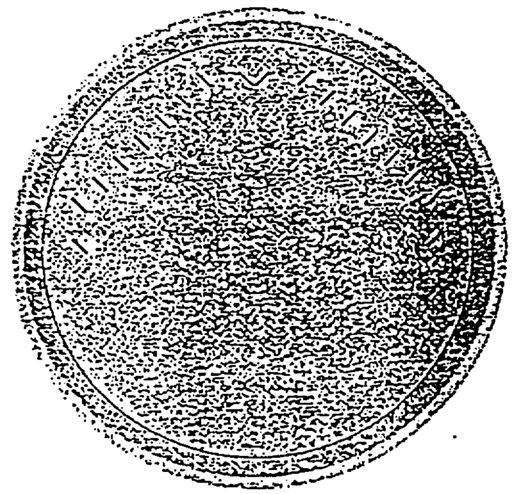
A complex combinatorial geometry (CG) was defined to model the Big Rock Point reactor vessel and internals. The neutron windows, thermal shield, seal weights, vessel cladding, vessel wall, seal housing, top guide, top guide plate, and steam baffle support were components considered in the model. The union and intersection of 47 different geometric bodies forming a total of 29 different zones accounted for these components. The validity of the geometric configuration was confirmed through multiple PICTURE[9] slices through the pressure vessel in different direction cosines. An X-Y (horizontal) cut through the upper section of pressure vessel shows the top guide bars in their open position, Figure 3-6 (upper right). An X-Y (horizontal) cut through the midsection of pressure vessel demonstrates the neutron window position, Figure 3-6 (lower right). A Y-Z (vertical) cut through the entire pressure vessel shows how the components are vertically situated in the model, Figure 3-6 (left).

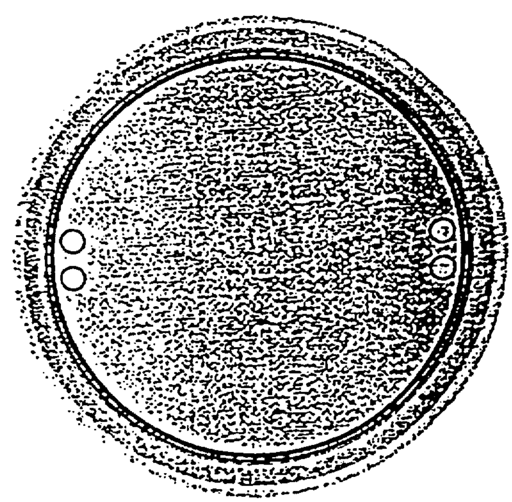
Figure 3-6

PICTURE Generated Images of Big Rock Point QAD-CGGP:

Y-Z vertical cut through the entire pressure vessel **(Left)**, X-Y horizontal cut through the upper section of pressure vessel **(Upper Right)**, X-Y horizontal cut through the midsection of pressure vessel **(Lower Right)**







With the QAD-CGGP combinatorial geometry verified, detectors were added to the model in the positions simulating actual survey position. The video made during the survey process confirms that the bottom of the detector was on contact with the top guide plate at the 609 foot survey location. All detector locations in the model were relative to this vertical contact point.

Multiple QAD-CGGP runs were then performed with different zones defined as source regions using the ANISN\ORIGEN2 activity results to define the Co-60 source strength. The analysis assumes that all measured doses at the time of surveys are from the 1.33 MeV and 1.17 MeV gammas emitted with each decay of Co-60. Source regions included the thermal shield, seal weights, top guide plate, and the steam baffle support.

3.2.2 MegaShield™ Models

Because of the relatively high activation levels of the top guide and neutron windows, the surrounding components had a negligible contribution to the survey results for these two components. Since the geometry of these components is relatively simplistic, MegaShield™ was used to model them.

Multiple detector locations were incorporated into the models to simulate actual survey points. The MegaShield™ models for the top guide and neutron windows are discussed separately below.

3.3 Co-60 Normalization Results

The survey dose rates were used in conjunction with QAD-CGGP to yield the Co-60 content for each component contributing to the dose rate. Once the Co-60 content for a component is calculated, the remaining curie content can be calculated based on scaling factors determined from the ANISN/ORIGEN2 activation analysis. The results are then decayed from the survey date to September 2002, which is the estimated shipping date of the reactor pressure vessel. All components with an activity greater than one curie were normalized to survey results.

3.3.1 Thermal Shield, Seal Weights, Vessel Clad, Wall and Insulation

The thermal shield normalization was performed in the active fuel region using the average of the two axial surveys on the south-southwest and south-southeast side of reactor vessel, see section 3.1.1. The thermal shield model also considered the following regions as source regions; neutron windows, top guide plate and seal weights. The source activity of

each QAD-CGGP source region was based on the Co-60 curie content as calculated by ORIGEN2 as of December 1998. Because the normalization was performed in the active fuel section of the thermal shield, the same active fuel section in which the radial flux models were developed, it is reasonable to apply the same normalization factor to other radial components; the seal weights, vessel cladding, wall and insulation. Table 3-1 shows the dose contribution from each component as a function of height.

Table 3-1

Dose Contribution to Surveys in Active Fuel Region

Survey Height (ft)	Thermal Shield	Seal Weights	Neutron Windows	Top Guide Plate	Total
603	96%	4%	0%	0%	100%
604	96%	4%	0%	0%	100%
605	96%	4%	0%	0%	100%
606	94%	4%	0%	2%	100%
607	81%	3%	0%	16%	100%

3.3.2 Top Guide Plate Normalization

The 609 foot and 610 foot measurements from the axial surveys represent contact and one foot measurements for the top guide plate. Both readings have over 98 percent of their dose contribution from the top guide plate, but the one foot location was chosen for the normalization because it "sees" more of the top guide plate activity. The top guide plate conservatively had the activity evenly distributed throughout the plate.

3.3.3 Top Guide Normalization

The top guide was analyzed using MegaShield™. The top guide was normalized based on the average dose of the top guide bars. Because the average dose was used, the ORIGEN2 activity was evenly distributed in the top guide bars. The detector orientation was identical to the actual survey location, i.e., with the 12 inch standoff in contact with two guide bars, centering the detector between the two guide bars. Three more detectors were then added to the Megashield™ model in 7.4 inch increments to simulate dose contribution from the surrounding grid bars. Because of symmetry, two grid bars contribute to the dose at each detector location. The total measured dose is then calculated and

normalized to the Co-60 content of the top guide grid bars. This method accurately calculates the total activity of the guide bars, and yields results for the average top guide grid bar.

3.3.4 Baffle Support Plate Normalization

A contact dose rate measurement at the 614 foot elevation was used to normalize the baffle support plate because of its low curie content.

3.3.5 Neutron Windows Normalization

The neutron windows were first analyzed using MegaShield™. The neutron windows were normalized based on the average dose of a set of two windows. The axial average of the dose rate was divided in two, because the two neutron windows were equidistant from the detector. Therefore, each neutron window contributed half the measured dose. For the MegaShield™ normalization, the ORIGEN2 activity was evenly distributed in the neutron windows. This produced overly conservative results, because it biased activity toward the far side of the neutron windows, away from the modeled detector location. This lowered the calculated dose rate and increased the normalized activity.

The same calculation was run using QAD-CGGP. The ORIGEN2 Co-60 was used as a QAD-CGGP source term. The results between MegaShield™ and QAD-CGGP were in close agreement, 7 percent. Then in QAD-CGGP to more accurately model the neutron windows, the activity was distributed azimuthally around the windows using a cosine distribution, because average thermal flux incident on the core side of the windows was approximately twice the flux incident on the far side of the neutron windows. Azimuthally distributing source reduced the normalized activity of the neutron windows by approximately 28 percent.

4.0 ESTIMATED COMPONENT RADIOACTIVITY AND CLASSIFICATION STATUS

The ANISN computer program results discussed in section 2.0 were used in conjunction with the normalized ORIGEN2 activation analysis results to determine activity and scaling factors as a function of component type and locale within the vessel. Section 3.0 describes the methods employed to normalize the ORIGEN2 results using radiation level measurements.

The estimated activity and the 10 CFR Part 61 classification status for each component presented in this section reflects the normalized results and should provide the basis for planning and regulatory compliance.

4.1 Overview

Table 4-1 summarizes the results in terms of total activity and the 10 CFR Part 61 Class C fractions. As shown, the estimated activity of all components as of September 1, 2002 is about 72,700 curies contained in the vessel and internals. Most of this activity, 48,400 curies, is contained within the GTCC Top Guide Bars. The table separates the components into three categories; GTCC material, Low Level Radioactive Waste (LLRW) internals components which can be shipped intact within the reactor vessel, and the reactor vessel.

4.2 Basis For Component Radioactivity

The normalized results for the individual components shown on Table 4-1 are discussed separately below. Weights and volumes are given for each component as well as the drawing reference from which the physical properties were determined. These results are based on the following assumptions.

Decay
Estimated activities were decay corrected to September 1, 2002, which is the earliest anticipated date to commence shipping operations.

Short Lived Activation Products
Activation products discussed herein include radionuclides with half-lives greater than 90 days. Because of the relatively long cooling time, short-lived activation product radionuclide activities (Co-58, Cr-51 and Fe-59, etc.) are not shown on the component specific results.

Surface Contaminants

The surface contamination on the internal surfaces exposed to primary coolant was determined by a direct sample that was obtained from the inlet baffle of the Big Rock Point Reactor Pressure Vessel. A ¼ inch drill bit was used to obtain a sampling of coupon metal shavings from the core spray nozzle. The 0.52 gram sample was then sent to Teledyne Brown Engineering Environmental Services for an isotopic analysis. A summary of the results from the analysis is given in Appendix B. Since the total neutron fluence on the core spray nozzle was extremely low because of its location, all measurable activity is conservatively assumed to be from surface contamination. Knowing the surface area and total curie content of the sample, the surface contamination is approximately 10.7 $\mu Ci/in^2$. The wetted surface area of each component, along with its total curie content and 10 CFR Part 61 classification including surface contamination is given in Table 4-1. For components with more than one curie of total activity the isotopic nuclide distribution is given in Tables 4-2 to 4-13.

TABLE 4-1
Estimated Component Activity and NRC Classification Status

	Drawing Numbers	Wetted Surface Area (ft²)	Volume (ft³)	Weight (lbs)	Total Curies	Co-60 Curies	Part 61 Table 1 Fraction	Part 61 Table 2 Fraction
GTCC Material								
Top Guide	197E861	2.05E+01	3.05E+00	1.53E+03	4.84E+04	2.12E+04	8.56	11.87
	Subtotal	**2.05E+01**	**3.05E+00**	**1.53E+03**	**4.84E+04**	**2.12E+04**		
LLRW Internals								
Steam Baffle	794E830	1.82E+02	3.27E+00	1.64E+03	1.09E+01	4.95E+00	0.02	< 0.01
Sparger	M248 Sh 46-1, 41-2	2.05E+01	2.60E-01	1.30E+02	8.78E-01	3.97E-01	0.03	< 0.01
Top Guide Plate	197E118	3.90E+1	1.63E+00	8.13E+02	1.17E+03	5.38E+02	0.41	0.54
Seal Housing	706E276	3.00E+1	2.15E+00	1.07E+03	1.45E+02	6.60E+01	0.04	0.05
Thermal Shield	197E853	4.14E+02	2.59E+01	1.30E+04	7.45E+03	3.83E+03	0.18	0.20
Thermal Shield Retainer (6 units)	237E956	1.39E+01	1.09E+00	5.47E+02	1.18E+02	5.94E+01	0.07	0.08
Seal Weights (12 units)	706E276, 104R175	2.59E+02	1.02E+01	5.08E+03	8.54E+02	5.89E+02	0.08	0.04
Neutron Windows (4 units)	107C353-9	4.48E+01	2.46E+00	1.23E+03	1.37E+04	6.92E+03	2.09	2.67
Core SOupport Plate	762-D229, E201-809-3	1.53E+02	5.08E+00	2.54E+03	1.4E-01	1.36E-01	0.01	< 0.01
Inlet Diffuser (2 units)	795E369	2.44E+01	3.97E-01	1.98E+02	2.24E-02	2.18E-02	0.02	< 0.01
Inlet Baffle	795E421	1.78E+02	4.64E+00	2.32E+03	1.63E-01	1.59E-01	0.01	< 0.01
	Subtotal	**1.36E+03**	**5.71E+01**	**2.85E+04**	**2.35E+04**	**1.20E+04**		
Reactor Vessel Assembly								
Reactor Vessel	197E853, 104R175, F 230-791-2	7.69E+02	4.14E+02	2.03E+05	7.98E+02	2.82E+02	< 0.01	< 0.01
Reactor Vessel Insulation	795E369	NA	2.59E+02	5.18E+03	5.21E+01	3.05E+01	< 0.01	< 0.01
	Subtotal	**7.69E+02**	**6.73E+02**	**2.08E+05**	**8.49E+02**	**3.13E+02**		
Reactor and Internals	**TOTAL**	**2.15E+03**	**7.33E+02**	**2.38E+05**	**7.27E+04**	**3.35E+04**		

4.2.1 GTCC Top Guide (DWG 197E861)

This component provides lateral alignment for the top of the fuel assemblies. It consists of a series of bars hinged at one end, which form a checkerboard pattern above the core. They are attached to the circular Top Guide Plate (see item 3 below) and weigh about 1,530 lbs.

These top guide bars contain a total of 48,400 curies and are well above Class C limits. Table 4-2 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status.

Table 4-2

Component: Top Guide		
Component Weight (lb):		1.53E+03
Total Activity Curies:		4.84E+04
Co-60 Activity Curies:		2.12E+04
Part 61 Table 1 C Fraction		8.56
Part 61 Table 2 C Fraction		11.87
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	7.40E-11	5.12E-05
C 14	1.44E-05	9.97E+00
Mn 54	2.21E-05	1.53E+01
Fe 55	2.89E-02	2.00E+04
Co 60	3.06E-02	2.12E+04
Ni 59	7.03E-05	4.87E+01
Ni 63	1.04E-02	7.17E+03
Zn 65	2.24E-11	1.55E-05
Nb 94	1.12E-07	7.79E-02
Tc 99	1.33E-08	9.19E-03
I 129	2.16E-11	1.49E-05
Cs 137 <LLD>	2.91E-11	2.01E-05
Ce 144 <LLD>	2.05E-12	1.42E-06
Pu 238	5.25E-13	3.64E-07
Pu 239/240	1.17E-12	8.10E-07
Am 241	2.28E-12	1.58E-06
Cm 242	3.53E-15	2.44E-09
Totals	6.99E-02	4.84E+04

4.2.2 LLRW Internals Components

The estimated total activity for the LLRW internals components are 23,500 curies. Individual component estimates are presented below. Note that where a component was found to contain less than one curie, a radionuclide distribution is not provided.

1. **Steam Baffle (DWG 794E830)**

   This component is located above the steam outlet nozzles. It consists of four plates attached with hinges to a circular ring. The unit is estimated to weigh about 1,640 lbs.

   The unit contains about 11 curies and is well below Class C limits. Table 4-3 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. This component can be shipped as LLRW within the intact reactor vessel. The steam baffle is Class A waste.

Table 4-3

Component: Steam Baffle		
Component Weight (lb):		1.64E+03
Total Activity Curies:		1.09E+01
Co-60 Activity Curies:		4.95E+00
Part 61 Table 1 C Fraction		0.02
Part 61 Table 2 C Fraction		< 0.01
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	6.14E-10	4.56E-04
C 14	2.82E-09	2.10E-03
Mn 54	3.83E-09	2.85E-03
Fe 55	5.70E-06	4.24E+00
Co 60	6.66E-06	4.95E+00
Ni 59	1.83E-08	1.36E-02
Ni 63	2.23E-06	1.66E+00
Zn 65	1.86E-10	1.38E-04
Nb 94	2.01E-11	1.50E-05
Tc 99	7.28E-11	5.41E-05
I 129	1.79E-10	1.33E-04
Cs 137 <LLD>	2.41E-10	1.79E-04
Ce 144 <LLD>	1.70E-11	1.27E-05
Pu 238	4.36E-12	3.24E-06
Pu 239/240	9.70E-12	7.21E-06
Am 241	1.89E-11	1.41E-05
Cm 242	2.93E-14	2.18E-08
Totals	1.46E-05	1.09E+01

2. Sparger (DWG M248, SH46-1, 41-2)

This component is located below and supported by the steam baffle. It is schedule 40, two inch diameter pipe with 36 spray nozzles attached to it. This component weighs about 130 lbs. and has an activity of about 1.0 curie.

Table 4-4 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. It is Class A waste and can be shipped as LLRW within the intact reactor vessel.

Table 4-4

Component: Sparger		
Component Weight (lb):		1.30E+02
Total Activity Curies:		8.78E-01
Co-60 Activity Curies:		3.97E-01
Part 61 Table 1 C Fraction		0.03
Part 61 Table 2 C Fraction		< 0.01
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	8.68E-10	5.12E-05
C 14	2.89E-09	1.70E-04
Mn 54	3.67E-09	2.16E-04
Fe 55	5.85E-06	3.45E-01
Co 60	6.72E-06	3.97E-01
Ni 59	1.87E-08	1.11E-03
Ni 63	2.29E-06	1.35E-01
Zn 65	2.63E-10	1.55E-05
Nb 94	1.94E-11	1.15E-06
Tc 99	1.02E-10	6.01E-06
I 129	2.53E-10	1.49E-05
Cs 137 <LLD>	3.41E-10	2.01E-05
Ce 144 <LLD>	2.41E-11	1.42E-06
Pu 238	6.16E-12	3.64E-07
Pu 239/240	1.37E-11	8.10E-07
Am 241	2.68E-11	1.58E-06
Cm 242	4.14E-14	2.44E-09
Totals	1.49E-05	8.78E-01

3. Top Guide Plate (DWG 197E118)

This component is a 1 inch thick circular plate with a core cutout section with an OD of 99.2 inches. It is attached to and supports the top guide. The unit is estimated to weigh about 813 lbs.

The unit contains a total of 1,170 curies and meets Class C limits. Table 4-5 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. This component can be shipped as LLRW within the intact reactor vessel.

Table 4-5

Component: Top Guide Plate		
Component Weight (lb):		8.13E+02
Total Activity Curies:		1.17E+03
Co-60 Activity Curies:		5.38E+02
Part 61 Table 1 C Fraction		0.41
Part 61 Table 2 C Fraction		0.54
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	2.65E-10	9.77E-05
C 14	6.22E-07	2.30E-01
Mn 54	1.11E-06	4.09E-01
Fe 55	1.25E-03	4.61E+02
Co 60	1.46E-03	5.38E+02
Ni 59	3.51E-06	1.29E+00
Ni 63	4.69E-04	1.73E+02
Zn 65	8.02E-11	2.96E-05
Nb 94	5.23E-09	1.93E-03
Tc 99	7.12E-10	2.63E-04
I 129	7.73E-11	2.85E-05
Cs 137 <LLD>	1.04E-10	3.84E-05
Ce 144 <LLD>	7.35E-12	2.71E-06
Pu 238	1.88E-12	6.94E-07
Pu 239/240	4.19E-12	1.55E-06
Am 241	8.17E-12	3.01E-06
Cm 242	1.26E-14	4.66E-09
Totals	3.18E-03	1.17E+03

4. Seal Housing (DWG 706E276)

This component is a 105.7-inch OD annular ring 3 inches high located at the top of the thermal shield between the reactor vessel wall and the thermal shield. The unit is estimated to weigh about 1,070 lbs.

The unit contains a total of 145 curies and is Class B waste. Table 4-6 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. This component can be shipped as LLRW within the intact reactor vessel.

Table 4-6

Component: Seal Housing		
Component Weight (lb):		1.07E+03
Total Activity Curies:		1.45E+02
Co-60 Activity Curies:		6.60E+01
Part 61 Table 1 C Fraction		0.04
Part 61 Table 2 C Fraction		0.05
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	1.54E-10	7.51E-05
C 14	5.75E-08	2.80E-02
Mn 54	7.39E-08	3.60E-02
Fe 55	1.16E-04	5.65E+01
Co 60	1.36E-04	6.60E+01
Ni 59	3.70E-07	1.80E-01
Ni 63	4.53E-05	2.21E+01
Zn 65	4.67E-11	2.27E-05
Nb 94	4.40E-10	2.14E-04
Tc 99	6.55E-11	3.19E-05
I 129	4.50E-11	2.19E-05
Cs 137 <LLD>	6.06E-11	2.95E-05
Ce 144 <LLD>	4.28E-12	2.08E-06
Pu 238	1.09E-12	5.33E-07
Pu 239/240	2.44E-12	1.19E-06
Am 241	4.75E-12	2.32E-06
Cm 242	7.36E-15	3.58E-09
Totals	2.97E-04	1.45E+02

5. Thermal Shield (DWG 197E853)

This component is a 92 inch high, 1.5 inch thick cylinder with an OD of 103 inches. The top of the Thermal Shield is 14 inches above the top of the 70 inch active core region and extends 8 inches below the active core region. The component weighs about 13,000 lbs. and contains about 7,450 curies.

Table 4-7 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. It is Class C waste and can be shipped as LLRW within the intact reactor vessel.

Table 4-7

Component: Thermal Shield		
Component Weight (lb):		1.30E+04
Total Activity Curies:		7.45E+03
Co-60 Activity Curies:		3.83E+03
Part 61 Table 1 C Fraction		0.18
Part 61 Table 2 C Fraction		0.20
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	1.76E-10	1.04E-03
C 14	2.22E-07	1.31E+00
Mn 54	7.77E-07	4.58E+00
Fe 55	4.41E-04	2.60E+03
Co 60	6.51E-04	3.83E+03
Ni 59	1.37E-06	8.04E+00
Ni 63	1.71E-04	1.01E+03
Zn 65	5.33E-11	3.14E-04
Nb 94	2.58E-09	1.52E-02
Tc 99	5.02E-10	2.95E-03
I 129	5.14E-11	3.03E-04
Cs 137 <LLD>	6.92E-11	4.07E-04
Ce 144 <LLD>	4.89E-12	2.88E-05
Pu 238	1.25E-12	7.36E-06
Pu 239/240	2.78E-12	1.64E-05
Am 241	5.43E-12	3.20E-05
Cm 242	8.40E-15	4.95E-08
Totals	1.26E-03	7.45E+03

6. Thermal Shield Retainer (DWG 237E956, 706E276)

This component consists of six bracket-like supports to which the thermal shield is attached. These brackets are welded to the vessel wall and bolted to the thermal shield. The unit is estimated to weigh about 547 lbs.

The unit contains a total of 118 curies and is Class B waste. Table 4-8 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. This component can be shipped as LLRW within the intact reactor vessel.

Table 4-8

Component: Thermal Shield Retainer		
Component Weight (lb):		5.47E+02
Total Activity Curies:		1.18E+02
Co-60 Activity Curies:		5.94E+01
Part 61 Table 1 C Fraction		0.07
Part 61 Table 2 C Fraction		0.08
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	1.40E-10	3.48E-05
C 14	8.56E-08	2.12E-02
Mn 54	2.54E-07	6.30E-02
Fe 55	1.71E-04	4.24E+01
Co 60	2.39E-04	5.94E+01
Ni 59	5.34E-07	1.33E-01
Ni 63	6.64E-05	1.65E+01
Zn 65	4.24E-11	1.05E-05
Nb 94	9.13E-10	2.27E-04
Tc 99	1.74E-10	4.33E-05
I 129	4.09E-11	1.01E-05
Cs 137 <LLD>	5.51E-11	1.37E-05
Ce 144 <LLD>	3.89E-12	9.65E-07
Pu 238	9.95E-13	2.47E-07
Pu 239/240	2.21E-12	5.49E-07
Am 241	4.32E-12	1.07E-06
Cm 242	6.68E-15	1.66E-09
Totals	4.78E-04	1.18E+02

7. Seal Weights (DWG 104R175, 706E276)

These 12 weights are located between the annulus formed by the thermal shield and the reactor vessel cladding. Each seal weight consists of three 1x18x27 inch plates. This group of weights contains about 854 curies and is estimated to weigh about 5,080 lbs.

Table 4-9 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. It is Class B waste and can be shipped as LLRW within the intact reactor vessel.

Table 4-9

Component: Seal Weights		
Component Weight (lb):		5.08E+03
Total Activity Curies:		8.54E+02
Co-60 Activity Curies:		5.89E+02
Part 61 Table 1 C Fraction		0.08
Part 61 Table 2 C Fraction		0.04
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	2.81E-10	6.47E-04
C 14	4.46E-08	1.03E-01
Mn 54	6.30E-07	1.45E+00
Fe 55	8.27E-05	1.91E+02
Co 60	2.55E-04	5.89E+02
Ni 59	2.21E-07	5.09E-01
Ni 63	3.11E-05	7.18E+01
Zn 65	8.50E-11	1.96E-04
Nb 94	1.37E-09	3.16E-03
Tc 99	4.16E-10	9.60E-04
I 129	8.19E-11	1.89E-04
Cs 137 <LLD>	1.10E-10	2.54E-04
Ce 144 <LLD>	7.79E-12	1.80E-05
Pu 238	1.99E-12	4.60E-06
Pu 239/240	4.44E-12	1.02E-05
Am 241	8.65E-12	2.00E-05
Cm 242	1.34E-14	3.09E-08
Totals	3.70E-04	8.54E+02

8. Neutron Windows (DWG 107C353-9)

These four units are 6 inch schedule 160 pipe with end caps and an overall height of 72 inches. They are filled with helium during operation and are located inside the thermal shield, in the active fuel height, on the east and west side of the reactor vessel. The four units contain about 13,700 curies and weigh about 1,230 lbs. including the mounting brackets.

Table 4-10 presents the radionuclide content and distribution for these components and the 10 CFR Part 61 status. These are high activity waste, and can be concentration averaged with the thermal shield in accordance with the NRC Branch Technical Position (BTP) to be shipped as LLRW within the intact reactor vessel. To minimize vessel exterior dose rates, these four units may have to be removed and placed in the center of the vessel for shipment.

Table 4-10

Component: Neutron Windows		
Component Weight (lb):		1.23E+03
Total Activity Curies:		1.37E+04
Co-60 Activity Curies:		6.92E+03
Part 61 Table 1 C Fraction		2.09
Part 61 Table 2 C Fraction		2.67
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	2.01E-10	1.12E-04
C 14	2.85E-06	1.59E+00
Mn 54	1.28E-05	7.14E+00
Fe 55	9.83E-03	5.48E+03
Co 60	1.24E-02	6.92E+03
Ni 59	1.78E-05	9.91E+00
Ni 63	2.33E-03	1.30E+03
Zn 65	6.09E-11	3.40E-05
Nb 94	2.84E-08	1.58E-02
Tc 99	4.62E-09	2.58E-03
I 129	5.87E-11	3.27E-05
Cs 137 <LLD>	7.91E-11	4.41E-05
Ce 144 <LLD>	5.58E-12	3.11E-06
Pu 238	1.43E-12	7.96E-07
Pu 239/240	3.18E-12	1.77E-06
Am 241	6.20E-12	3.46E-06
Cm 242	9.60E-15	5.35E-09
Totals	2.46E-02	1.37E+04

9. <u>Core Support Plate (DWG E201-809-3, 762-D229)</u>

This component is a 1.5 inch thick square plate having rounded corners 70.5 inches wide with a series of large diameter flow holes. This component weighs about 2,540 lbs. and has an estimated activity of less than one curie. It is very low activity Class A waste.

10. <u>Inlet Diffuser (DWG 795E369)</u>

These two components are 1 inch thick plates with two curved, 3/8 inch thick pieces of steel used to redirect the flow of water from the cold leg of plant laterally to help water flow more easily through the inlet baffle. These components weigh about 198 lbs. and have an estimated total, combined activity of much less than one curie. They are very low activity Class A waste.

11. <u>Inlet Baffle (DWG 795E421)</u>

This component is a hexagonal parallel piped hollow skirt, approximately 3 feet high, constructed of 5/8 inch thick steel. It directs water flow upward toward the core of the reactor. This component weighs about 2,320 lbs. and has an estimated activity of less than one curie. It is very low activity Class A waste.

4.2.3 Reactor Vessel Assembly

The reactor vessel assembly consists of the carbon steel body and with stainless steel cladding, and the insulation surrounding the body and the head.

1. <u>Reactor Vessel Head (DWG 197E853)</u>

The reactor vessel head is a 5.25 inch thick carbon steel hemisphere with a flanged section that bolts to the lower body section of the reactor pressure vessel. The head itself is fabricated of carbon steel with a 5/16 inch thick stainless steel cladding on the interior. The vessel head, including the cladding, weighs about 40,200 lbs and contains less than one curie. The head is low activity Class A waste.

The reactor vessel head has approximately 1,060 lbs of insulation on the exterior. A detailed description of the insulation is given in section 3 below. The insulation on the head has much less than one curie and is low activity Class A waste.

2. Reactor Vessel and Cladding (DWG 197E853)

The reactor vessel and cladding comprised two distinct regions in the radial model since they are fabricated of carbon steel and stainless steel, respectively. The specific activity within the cladding is considerably higher than the vessel because of its proximity to the active core and elemental composition of the stainless steel.

The reactor vessel body, including the cladding, weighs about 162,220 lbs. The 5.25-inch thick carbon steel wall contains about 589 curies and the 5/16 inch thick stainless cladding contains about 208 curies.

Tables 4-11 and 4-12 present the radionuclide content and distribution for these components and the 10 CFR Part 61 status. The tables contain the total weight including the vessel head, which is the intended shipping configuration.

The surface contaminants are assumed to all be on the vessel cladding that is broken out in its own Table 4-11. This component is Class A waste.

Table 4-11

Component: Vessel Cladding (including head)		
Component Weight (lb):		5.03E+03
Total Activity Curies:		2.09E+02
Co-60 Activity Curies:		1.30E+02
Part 61 Table 1 C Fraction		0.04
Part 61 Table 2 C Fraction		0.01
Nuclide	Curies/g	Estimated Curies
H 3 <LLD>	8.44E-10	1.92E-03
C 14	1.29E-08	2.95E-02
Mn 54	1.18E-07	2.69E-01
Fe 55	2.48E-05	5.66E+01
Co 60	5.70E-05	1.30E+02
Ni 59	7.15E-08	1.63E-01
Ni 63	9.47E-06	2.16E+01
Zn 65	2.56E-10	5.83E-04
Nb 94	2.78E-10	6.35E-04
Tc 99	1.69E-10	3.85E-04
I 129	2.46E-10	5.62E-04
Cs 137 <LLD>	3.32E-10	7.56E-04
Ce 144 <LLD>	2.34E-11	5.34E-05
Pu 238	5.99E-12	1.37E-05
Pu 239/240	1.33E-11	3.04E-05
Am 241	2.60E-11	5.94E-05
Cm 242	4.03E-14	9.18E-08
Totals	9.14E-05	2.09E+02

Table 4-12

Component: Vessel Wall (including head)		
Component Weight (lb):		1.98E+05
Total Activity Curies:		5.90E+02
Co-60 Activity Curies:		1.52E+02
Part 61 Table 1 C Fraction		< 0.01
Part 61 Table 2 C Fraction		< 0.01
Nuclide	Curies/g	Estimated Curies
C 14	3.89E-10	3.49E-02
Mn 54	8.01E-08	7.20E+00
Fe 55	4.72E-06	4.24E+02
Co 60	1.69E-06	1.52E+02
Ni 59	4.36E-10	3.92E-02
Ni 63	7.69E-08	6.90E+00
Nb 94	2.49E-11	2.24E-03
Tc 99	7.86E-11	7.06E-03
Total	6.57E-06	5.90E+02

3. **Exterior Insulation**

The reactor vessel insulation is comprised of stainless steel plate with a folded stainless steel foil between the plates. The net thickness of the foil is approximately 3 inches, and weighs 5,180 lbs. including the head insulation. The results also indicate that the stainless steel based insulation around the vessel exterior contains about 52 curies.

Table 4-13 presents the radionuclide content and distribution for this component and the 10 CFR Part 61 status. This component is Class A waste.

Table 4-13

Component: Vessel Insulation (including head)		
Component Weight (lb):		5.18E+03
Total Activity Curies:		5.21E+01
Co-60 Activity Curies:		3.05E+01
Part 61 Table 1 C Fraction		0.00
Part 61 Table 2 C Fraction		0.00
Nuclide	Curies/g	Estimated Curies
C 14	3.39E-09	7.97E-03
Mn 54	2.33E-08	5.49E-02
Fe 55	6.60E-06	1.55E+01
Co 60	1.30E-05	3.05E+01
Ni 59	1.96E-08	4.62E-02
Ni 63	2.54E-06	5.97E+00
Nb 94	6.00E-11	1.41E-04
Tc 99	1.44E-11	3.38E-05
Total	2.22E-05	5.21E+01

5.0 ESTIMATED ERRORS ASSOCIATED WITH THE SOURCE TERM CALCULATIONS AND NORMALIZATION

The estimated errors associated with the shielding analysis, source term and source normalization originate from several sources. Major areas of uncertainty include the error associated with the ANISN/ORIGEN2 activation methodology, cobalt impurity levels in the base metal, the point kernel shielding methodology and the detector response and location. Normalization of the source terms relative to measured dose rates limits the areas of uncertainty to the point kernel modeling and radiation detection, which are discussed below.

- **Radiation Detector**
  The potential errors for the measured survey results include instrument calibration and variances in the detector distance to the thermal shield and other internal components. The survey instrument used was a Dositec PR-2 high range probe. The Palisades Nuclear Power Plant Health Physics Procedure literature indicates that the detector accuracy with the high range probe is ±10%. When the standoff or detector was in contact with the component being survey, ±10% is error associated with the measurement (top guide bars, neutron windows, etc.). Changes in the distance between the detector and component being surveyed (top guide bars), because of standoff/detector location were taken into account in the QAD-CGGP and MegaShield™ models, but because the standoff was in contact with the component being surveyed, estimated error with these measurements was also 10%. However, the error associated with some measurements was greater than 10%. Using the top guide plate as an example of the worse case scenario (because the detector cable markings and not the standoff were used to measure axial height), the radiation survey measurements were taken at a distance of 12 inches "±0.5 inch" above the top guide plate. This 0.5 inch variance in detector location could cause an additional 9 percent variation in the detector response under water as determined from QAD-CGGP models. Since these errors are additive, the maximum error associated with the measured survey results is 19%.

- **Point Kernel Modeling**
  Detailed point kernel models were developed with QAD-CGGP to model the reactor vessel and internals as accurately as possible including local variations within the source region. However, there is error inherent in the use of the point kernel methodology itself. The point kernel technique can result in grossly underestimated dose results if the integration parameters are not sufficiently large. This is especially true when dealing with relatively close detector locations. There have been published guidelines regarding integration parameters for point kernel applications[10] and these were adhered to in the analysis. A study

regarding uncertainties in irradiated hardware characterization, NUREG/CR-4968[11], estimated the uncertainty for point kernel shielding techniques at ±25% which is reasonable for this analysis.

- **Normalization Error**

  As discussed above in section 3.3, the analysis results were normalized to the measured dose rate. The errors associated with the normalization factor used can be directly applied to all the shielding analysis results. The normalization factor was calculated as follows:

$$XNF = \text{Measured dose} / \text{Calculated dose}$$

The two parameters in this equation each have a maximum error associated with them as discussed above. The relative errors for the measured and calculated dose rates are 10-19% and 25% respectively. The dose rates and error used below are associated with the top guide bars, where the standoff is in contact (measured dose ±10%) with the component being surveyed.

Measured dose = 16,920 ±10%

Calculated dose = 28,400±25%

The total relative error for the calculated normalization factor (XNF) is given by the root sum of the squares as 27 % for calculation where the detector or standoff is on contact the component. For calculations where the detector distance is estimate at 12 inches ±0.5 inches (i.e. the top guide plate) the measure dose is known ±19% yielding a relative error of 31%. Therefore, it is reasonable to assume the results of the analysis are accurate within ±31%.

The 31% error associated with the transport and normalization of the Big Rock Point reactor pressure vessel and internals is reasonable for a three-dimensional synthesized flux calculation from one-dimensional radial and axial models. This is because the error associated with a full-blown three-dimensional transport calculation alone, as estimated by LEPRICON[12] (Least-squares EPRI CONsolidation), is 30%. The error in any transport calculations can be attributed to uncertainty in five areas: cross sections, source terms, the reactor model, calculational procedures (numeric differencing schemes), and the methods to setup and solve the transport model.

6.0 REFERENCES

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4. DLC-76, SAILOR-Coupled, Self-Shielded, 47-Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors, Oak Ridge National Laboratory Radiation Shielding Information Center, April 1991.

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6. DOE/RW 0184 Appendix 2A, Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, U.S. Department of Energy, December 1987.

7. ORNL/TM-11018, Standard-and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code, Ludwig, S. B., and Renier, J. P., December 1989.

8. Palisades Nuclear Power Plant Health Physics Procedure, Procedure No. HP 9.72 Revision 2.

9. CCC-545, SCALE 4.3 Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations for Workstations and Personal Computers, September 1995.

10. Nuclear Technology, "The Selection of Fixed-Order Quadratures in Point-Kernel Shielding Calculations", O.J. Wallace, American Nuclear Society, La Grange Park, Illinois, 1996, Vol. 113, pp.112-122.

11. NUREG/CR-4968, "Evaluation of Uncertainties in Irradiated Hardware Characterization", A. Levin, et al., Waste Management Group, Inc., Crompond, New York, October 1987.

12. Methodologies for Particle Transport Simulation of Nuclear Systems, "Sensitivity Analysis and Uncertainty Estimation" A. Haghighat, et al., Penn State University, University Park, PA, 1999 pp. 2-24.

APPENDICES

APPENDIX A

SURVEY PLAN

SURVEYS RESULTS

Big Rock Point Characterization Survey Plan

1.0 Overview

Part of the scope of work for the reactor pressure vessel (RPV) and internals characterization entails obtaining detailed radiation profiles on the reactor vessel internal components. These survey results will be used to benchmark and normalize the activation analysis results obtained from ANISN/ORIGEN2 calculations. The results will significantly decrease the uncertainties associated with the activation analysis calculations and help to ensure that the shielding capabilities for the planned shipping configurations are adequate. The survey results will be used in conjunction with a series of detailed QAD-CGGP and/or MORSE-CG shielding calculations to normalize the activation results.

Once the activation results are normalized to measured results they can be used as final characterization results for shipment and disposal of the reactor vessel and internals. These activation analysis results will also be supplemented with sample analysis results which are representative of the surface contaminants on the internals for final characterization.

There are three main areas of concern regarding the distribution of activity within the RPV and internals including the axial, azimuthal and radial distributions. In addition to these areas of concern, detailed surveys must be obtained on components, which are potentially greater than "Class C", material (GTCC) including the top fuel guide and neutron windows. Separate surveys will be required to provide empirical data for each of these areas as follows:

- Axial distribution of activity
- Azimuthal distribution of activity
- Radial distribution of activity
- Activation level of the top fuel guide
- Activation of the neutron windows.

This survey plan can be modified in the field with the concurrence of both the WMG and Consumers Energy Project Managers. The azimuthal distribution surveys can be eliminated in their entirety if it is found to be infeasible to obtain accurate survey locations and results.

2.0 Prerequisites

In order to provide access to the lower internal components, the steam baffle plates and the top fuel guide will need to be in their open positions. Having the top fuel guide open, will also minimize its contribution to the survey results for the south side of the

internals shroud. Water clarity will be required in order to identify and verify survey probe locations for the surveys.

An RWP will be required to perform the surveys and the radiation protection technician to perform the surveys should be qualified on both the PR-2 and the RO-7.

3.0 Equipment

Under water survey meters and probes will be required to perform the surveys. An Eberline RO-7 or Dositec PR-2 will suffice, with the PR-2 preferred due to its high range capability of up to 50,000 R/hr. The high-range, mid-range and low-range probes may be required for both the RO-7 and PR-2. All meters/probes to be used must have current calibration documentation on-file at BRP. The probe cables should be clearly marked in 1 foot increments to aid in axial placement of the probe relative to the survey locations. The 1 foot reference marks will provide fixed reference points relative to the surface of the water.

Standoff fixtures for the probes will be required since the top fuel guide is expected to have contact dose rates in excess of 50,000 R/hr. A 12 inch diameter circular standoff for use with the PR-2 is ideal. A graduated rectangular standoff for the RO-7 is desirable with probe locations at 6 inch increments up to a total 2 foot distance. Note the RO-7 will only be used as a contingency if problems are experienced with the PR-2. Standoff readings will be taken for all normalization survey results. Contact readings will also be obtained on some critical components and these will be used for information only.

A view box will be required to position the probe at the requisite underwater locations. A long handled pole with a J-hook or other attachment may be required to position the probe for some survey locations.

4.0 Radiation Survey Requirements

Detailed survey requirements including survey maps and data sheets to record radiation survey results are discussed for each of the required surveys below. Please note that additional surveys may be requested by the WMG project manager provided they are performed in accordance with plant procedures and in an ALARA fashion.

4.1 Axial Distribution Surveys

Radiation profiles will be obtained along the inner surface of the RPV and internals at the south side of the RPV (away from the open top guide) at the SSE and SSW locations as shown in Figure 1 (DWG # 104R175 Sh. 3). Radiation levels will be obtained using the PR-2 with the standoff at 1 foot axial increments and recorded on the survey form in Exhibit 1.

4.2 Azimuthal Distribution Surveys

Radiation levels will be obtained at six different azimuthal locations shown on Figure 2, at the 604' elevation and at the same azimuthal locations for the 606' and 608' elevations using the PR-2 with the standoff. A total of 18 radiation level readings will be recorded on the survey form shown in Exhibit 2. Note these will be the most difficult readings to obtain due to obstructions above this region. A J-hook may be required to position the probe in place.

4.3 Radial Distribution Surveys

The dose rates for the top fuel guide will be the most representative of the radial distribution of the neutron flux within the BRP core. Therefore, a profile will be obtained along the length of either of the open top fuel guide sections with the PR-2 and standoff. Dose rate measurements will start at the bottom center of the top fuel guide at the 610' elevation and readings will be taken at 6 inch axial increments. Note that the standoff should be in contact with the top fuel guide for all measurements obtained. Survey results may be recorded on BRP standard survey forms.

4.3 Top Fuel Guide Surveys

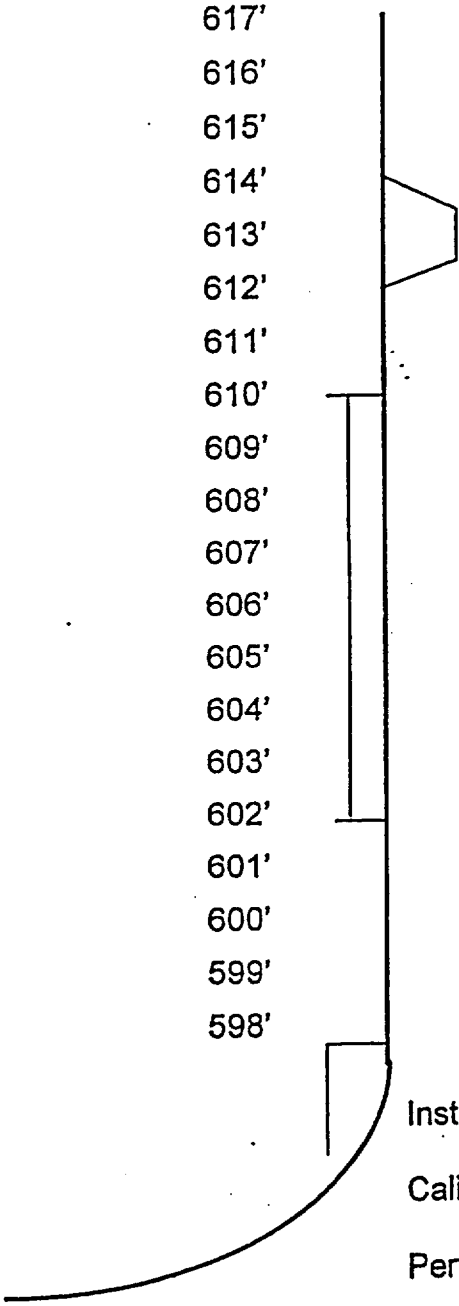
In addition to the surveys discussed above for the radial distribution, contact surveys of the top fuel guide are to be obtained to the extent practicable within the maximum range of the PR-2. A detailed survey is to be performed to define the maximum contact reading as well as a general distribution of contact readings. Readings should be the highest near the center (in its installed position) and decrease radially outward. Contact radiation levels should also be obtained on contact with the top fuel guide plate located around the periphery of the core region. The top fuel guide plate surveys should be obtained on the south side to minimize background from the top fuel guide. Survey results may be recorded on BRP standard survey forms.

4.4 Neutron Window Surveys

The neutron windows are another primary component of interest since they are potentially GTCC material. Radiation profiles should be obtained with the PR-2 with the standoff in contact with both neutron windows in a group. The neutron windows are in groups of 2 as shown in Figure 2 (DWG # 104R175 Sh. 4). Survey readings will be taken at 6 inch axial increments starting at the bottom (602' elevation) and continuing up to the top of the neutron windows (608' elevation). Contact readings may also be taken for additional information if time permits. Survey results may be recorded on BRP standard survey forms.

<u>Exhibit 1</u>

BRP RPV Characterization Survey Form – 1: Axial Distribution



Elevation View RPV	Survey Results R/hr South (180)	SouthEast
617'	617A ____________	617B____________
616'	616A ____________	616B____________
615'	615A ____________	615B____________
614'	614A ____________	614B____________
613'	613A ____________	613B____________
612'	612A ____________	612B____________
611'	611A ____________	611B____________
610'	610A ____________	610B____________
609'	609A ____________	609B____________
608'	608A ____________	608B____________
607'	607A ____________	607B____________
606'	606A ____________	606B____________
605'	605A ____________	605B____________
604'	604A ____________	604B____________
603'	603A ____________	603B____________
602'	602A ____________	602B____________
601'	601A ____________	601B____________
600'	600A ____________	600B____________
599'	599A ____________	599B____________
598'	598A ____________	598B____________

Instrument Used:__________ S/N:__________

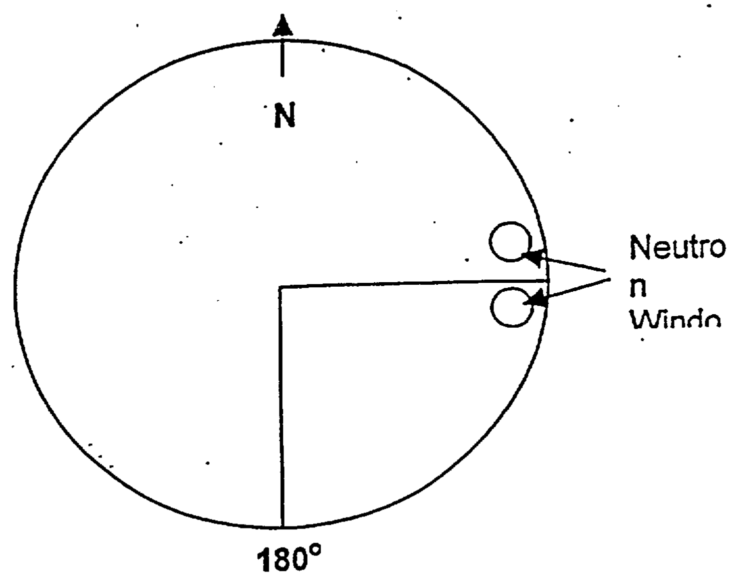
Calibration Due:__________ Standoff Inches____

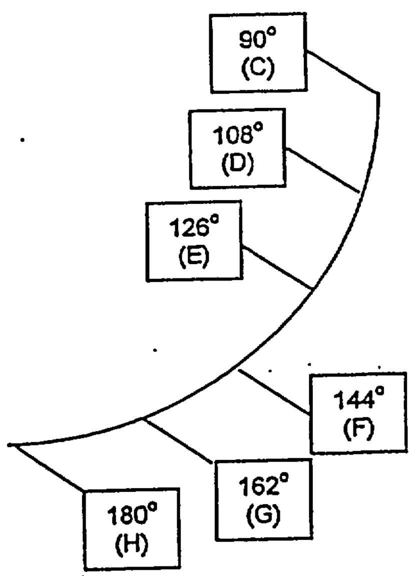
Performed By:__________ Date:_________

Reviewed By:__________ Date:_________

Exhibit 2

BRP Thermal Shield Characterization Survey Form – 2: Azimuthal Distribution
(Six measurements 18° apart with approximately
31 linear inches between survey locations)





	604' (R/hr)	606' (R/hr)	608' (R/hr)
C	______	______	______
D	______	______	______
E	______	______	______
F	______	______	______
G	______	______	______
H	______	______	______

Instrument Used:__________ S/N:__________

Calibration Due:__________ Standoff Inches____

Performed By:__________ Date:__________

Reviewed By:__________ Date:__________

Figure 1

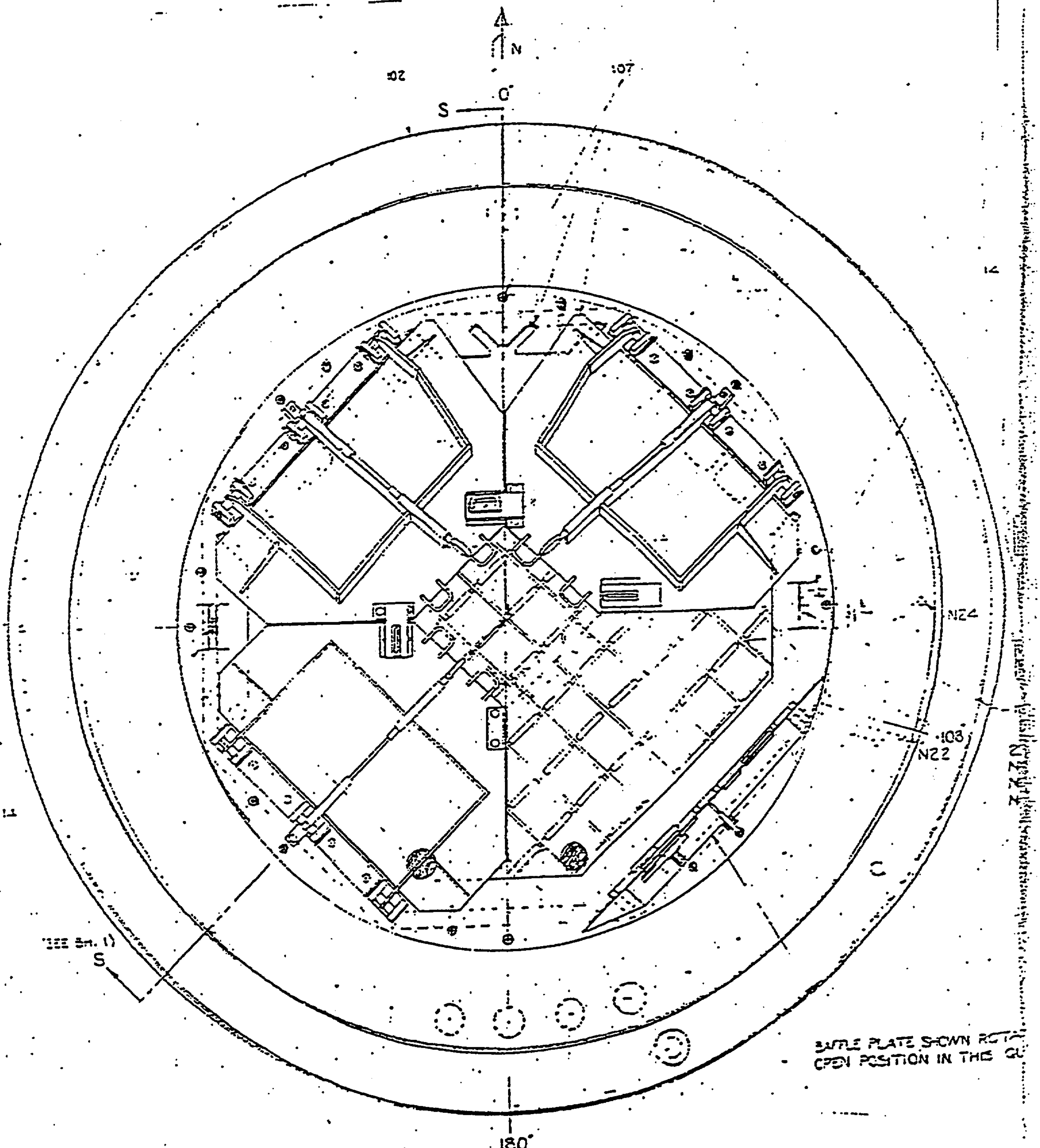
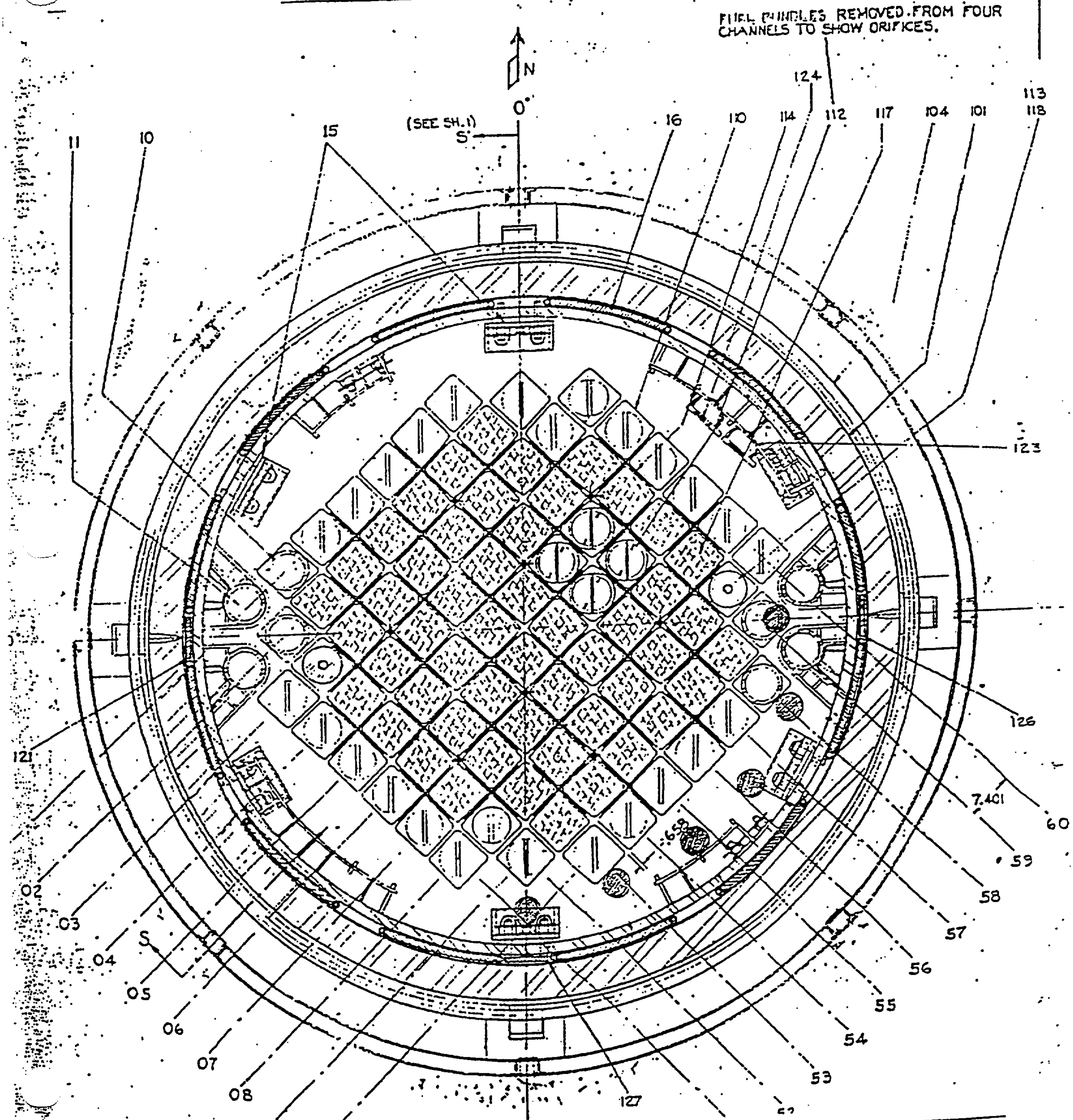

N
0°
S
N24
N22
S
180°
BAFFLE PLATE SHOWN ROTAT
OPEN POSITION IN THE GU

Figure 2



Big Rock Point RPV and Internals Radiation Survey Plan

RADIOLOGICAL SURVEY AND STATUS SHEET

-29-1 Rev.54

UFI: 24*11*01

/Time: 12/16/98 0955-1305 Technician: M. STEVENS K. TUITE D. SHEDLOCK	Location: Rx DECK / Rx VESSEL
Dose Rate: mRem/hr(γ) mRem/hr(β)	General Dose Rate: mRem/hr(γ) mRem/hr(β)
High Contamination: dpm/100cm²	General Contamination: dpm/100cm²
Airborne Concentration: µCi/cc	MD Spectrum No:

Area Posting: ☐ Contamination Area (CA) ☐ High Radiation Area (HRA) ☐ High Contamination Area (HCA) ☐ Radiation Area (RA) ☐ Locked High Radiation Area (LHRA) ☐ Airborne Area (ABN)

NOT REQUIRED

Special Instructions: ☐ Notify Rad Pro PRIOR to any "Work Entry" ☐ Air Sampling Required for Entry ☐ Notify Control Room When Entering and Exiting ☐ Survey Meter Required in HRA's ☐ High Rad Door Watch Required

1. ____ dpm/100cm²	5. ____ dpm/100cm²	9. ____ dpm/100cm²	13. ____ dpm/100cm²	17. ____ dpm/100cm²
2. ____ dpm/100cm²	6. ____ dpm/100cm²	10. ____ dpm/100cm²	14. ____ dpm/100cm²	18. ____ dpm/100cm²
3. ____ dpm/100cm²	7. ____ dpm/100cm²	11. ____ dpm/100cm²	15. ____ dpm/100cm²	19. ____ dpm/100cm²
4. ____ dpm/100cm²	8. ____ dpm/100cm²	12. ____ dpm/100cm²	16. ____ dpm/100cm²	20. ____ dpm/100cm²

Comments: Rx VESSEL CHARACTERIZATION SURVEYS

Map Key: # : General Area Dose Rate C/GA: Contact and General Area Dose Rate ⓢ: Smear Location *: Hot Spot ..x..x..: Radiological Boundary LDWA : Low Dose Waiting Area

Rx VESSEL CHARACTERIZATION SURVEYS WERE PERFORMED USING A DOSITECH PR-2 SURVEY METER WITH A HIGH RANGE PROBE. BORROWED FROM PALISADES. OPERATION OF THE PR-2 WAS IN ACCORDANCE WITH PALISADES PROCEDURES HP 9.72 AND HP 9.45. A 6" STANDOFF (12" DIA DISC) WAS USED TO OBTAIN UNIFORM DISTANCES FROM Rx VESSEL INTERNALS.

THE FOLLOWING SURVEYS ARE INCLUDED;

PG 2 - AXIAL DISTRIBUTION SURVEY OF Rx VESSEL
PG 3 - AZIMUTHAL DISTRIBUTION SURVEY OF Rx VESSEL GRID BARS
PG 4 - DOSE PROFILE OF NUETRON WINDOWS (WEST)
PG 5 - DOSE PROFILE OF

Unposted Hot Spots May be Present

Op's Rounds
☐ Gloves for Reach ins SEE RWP
☐ Class A
☐ Class B
☐ Class C

RP Surveys
☐ Gloves for Reach ins SEE RWP
☐ Class A
☐ Class B
☐ Class C

Tours/Inspections
☐ Gloves for Reach ins SEE RWP
☐ Class A
☐ Class B
☐ Class C

Work Entries
☐ Class A SEE RWP
☐ Class B
☐ Class C

Rate Meter SN: PR-2 #469	Cal Due: 11-3-99	Air Sampler: N/R	Cal Due: N/R
ting Inst. SN: N/R	Cal Due: N/R	Inst. Type SN: N/R	Cal Due: N/R
Performed By: Stevens / Tuite / Shedlock	Date: 12/16/98	Reviewed By:	Date:

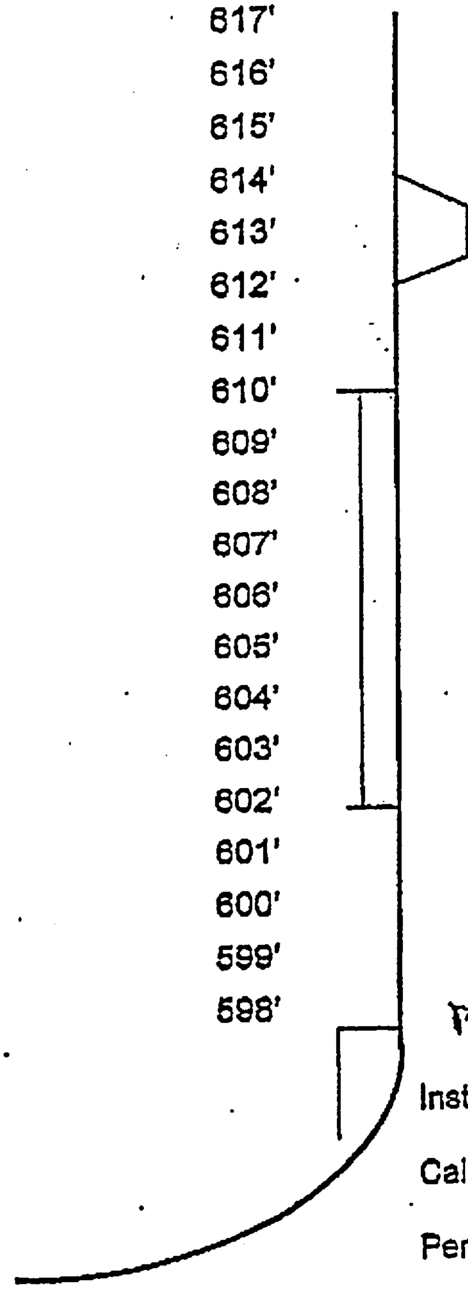
UFI: 24*11*01

RP-29-1

Exhibit 1

BRP RPV Characterization Survey Form – 1: Axial Distribution

Elevation View RPV



Elevation	Survey Results R/hr (2) South (180)		SouthEast	
617'	617A	< LLD	617B	12.5
616'	616A	25	616B	18.75
615'	615A	12.5	615B	18.75
614'	614A (1)	12.5 (contact)	614B (1)	25 (contact)
613'	613A	12.5	613B	37.5
612'	612A	12.5	612B	43.75
611'	611A	37.5	611B	112.5
610'	610A	275	610B	312.5
609'	609A (1)	1625 (contact)	609B (1)	8750 (contact)
608'	608A	2500	608B	5000
607'	607A	1500	607B	1500
606'	606A	1750	606B	1625
605'	605A	1375	605B	1500
604'	604A	1250	604B	1250
603'	603A	1000	603B	1062.5
602'	602A	625	602B	437.5
601'	601A	500	601B	100
600'	600A	625	600B	25
599'	599A	37.5	599B	25
598'	pms ~~598A~~ 598.5A (1)	25 (contact)	~~598B~~ 598.5B (1)	37.5 (contact)

Instrument Used: PR-2 S/N: # 469

Calibration Due: 11-3-99 Standoff Inches 6" (12" Dia

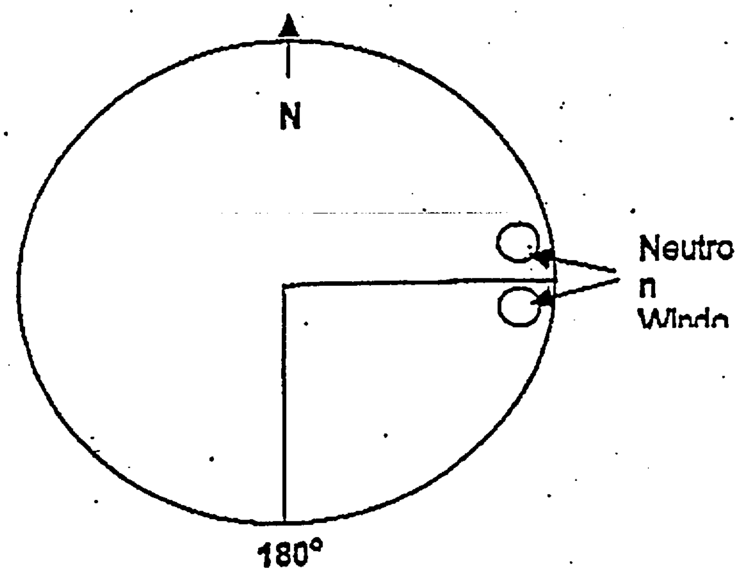
Performed By: M. Stevens K. Tuite Date: 12/16/98

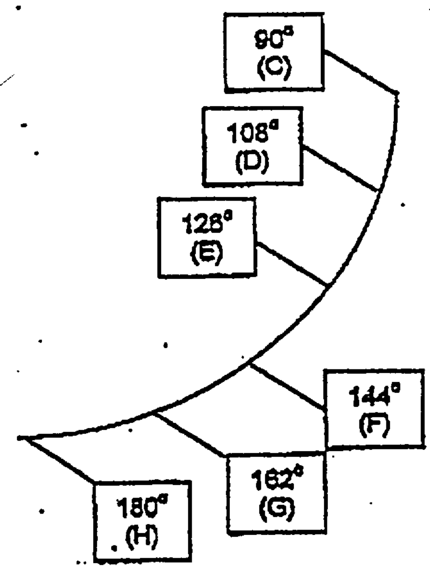
Reviewed By: ______ Date: ______

NOTES: (1) contact readings – all other readings taken at 6" dista using a 12" dia standoff.
(2) All readings have been corrected to Co-60 using a corr. factor of 1.25 as per Dal PR-2 procedure HP

Exhibit 2

BRP Thermal Shield Characterization Survey Form – 2: Azimuthal Distribution
(Six measurements 18° apart with approximately
31 linear inches between survey locations)





	604' (R/hr) ②	606' (R/hr) ②	608' (R/hr)
C	NOT TAKEN ③	NOT TAKEN ③	NOT TAKEN ③
① D	12,500	15,000	
① E	3125	3125	
① F	1500	1625	
① G	1500	2125	
① H	1500	1375	

Instrument Used: PR-2 S/N: 469

Calibration Due: 11-3-99 Standoff Inches 6" (12" Dia)

Performed By: M. Stevens / D. Shedlock Date: 12-16-98

Reviewed By: ______ Date: ______

NOTES:
① All readings taken at 6" distance using 12" dia. standoff.
② All readings have been corrected to Co-60 using a corr Factor of 1.25 as per Palisades Procedure HP 9.72

③ Readings not taken due to not being physically able to reach this l

RADIOLOGICAL SURVEY AND STATUS SHEET

P-29-1 Rev.54 UFI: 24*11*01

Date/Time: 12/16/98/1230	Technician: M. Stevens K. Tuite
Location: Rx Deck (Rx Vessel)	
High Dose Rate: 22,500 R/hr ~~mRem/hr~~(γ) N/R mRem/hr(β)	General Dose Rate: 20,000 R/HR ~~mRem/hr~~(γ) N/R mRem/hr(β)
High Contamination: N/R dpm/100cm²	General Contamination: N/R dpm/100cm²
Airborne Concentration: N/R μCi/cc	ND Spectrum No: N/R

Area Posting: N/A
- ☐ Contamination Area (CA)
- ☐ High Radiation Area (HRA)
- ☐ High Contamination Area (HCA)
- ☐ Radiation Area (RA)
- ☐ Locked High Radiation Area (LHRA)
- ☐ Airborne Area (ABN)

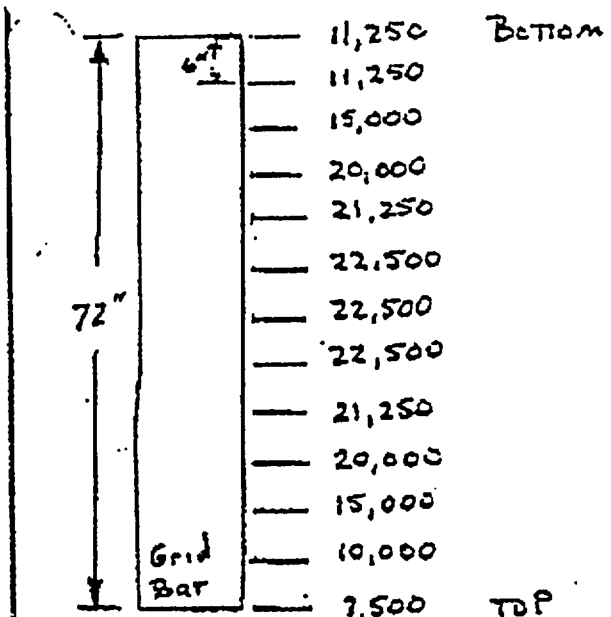
Special instructions: N/A
- ☐ Notify Rad Pro PRIOR to any "Work Entry"
- ☐ Air Sampling Required for Entry
- ☐ Notify Control Room When Entering and Exiting
- ☐ Survey Meter Required in HRA's
- ☐ High Rad Door Watch Required

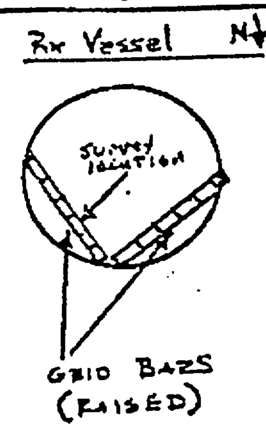
1. N/R dpm/100cm²	5. N/R dpm/100cm²	9. N/R dpm/100cm²	13. N/R dpm/100cm²	17. N/R dpm/100cm²
2. dpm/100cm²	6. dpm/100cm²	10. dpm/100cm²	14. dpm/100cm²	18. dpm/100cm²
3. dpm/100cm²	7. dpm/100cm²	11. dpm/100cm²	15. dpm/100cm²	19. dpm/100cm²
4. dpm/100cm²	8. dpm/100cm²	12. dpm/100cm²	16. dpm/100cm²	20. dpm/100cm²

Comments: Survey comming up between 3rd & 4th Gridbars (ve) in 6" increments using a 6" standoff (12" dia). All readings have been corrected to Co-60 using a corr. factor of 1.25 as per Palisades procedure HP 9.72

Map Key: # : General Area Dose Rate C/GA: Contact and General Area Dose Rate ⓢ: Smear Location --X--X-: Radiological Boundary LDWA : Low Dose Waiting Area *: Hot Spot

Dose Rates (R/HR)





Unposted Hot Spots May be Present

Op's Rounds
- ☐ Gloves for Reach Ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

RP Surveys
- ☐ Gloves for Reach Ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

Tours/Inspections
- ☐ Gloves for Reach Ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

Work Entries
- ☐ Class A SEE RWP
- ☐ Class B
- ☐ Class C

Dose Rate Meter SN: PR-2 #969	Cal Due: 11-3-99	Air Sampler: N/R	Cal Due: N/R
Counting Inst. SN: N/R	Cal Due: N/R	Inst. Type SN: N/R	Cal Due: N/R
Performed By: M. Stevens K. Tuite	Date: 12/16/98	Reviewed By: [signature]	Date:

RADIOLOGICAL SURVEY AND STATUS SHEET

-29-1 Rev.54 UFI: 24*11*01

Date/Time: 12/16/98 1300 · Technician: M. Stevens, K. Tuite	Location: Rx Deck - Rx Vessel
Dose Rate: mRem/hr(γ) N/R mRem/hr(β)	General Dose Rate: mRem/hr(γ) N/R mRem/hr(β)
High Contamination: N/R dpm/100cm²	General Contamination: N/R dpm/100cm²
Airborne Concentration: N/R µCi/cc	ND Spectrum No: N/R

Area Posting: N/A
- ☐ Contamination Area (CA)
- ☐ Radiation Area (RA)
- ☐ Airborne Area (ABN)
- ☐ High Radiation Area (HRA)
- ☐ Locked High Radiation Area (LHRA)
- ☐ High Contamination Area (HCA)

Special instructions: N/A
- ☐ Notify Rad Pro PRIOR to any "Work Entry"
- ☐ Survey Meter Required in HRA's
- ☐ Air Sampling Required for Entry
- ☐ High Rad Door Watch Required
- ☐ Notify Control Room When Entering and Exiting

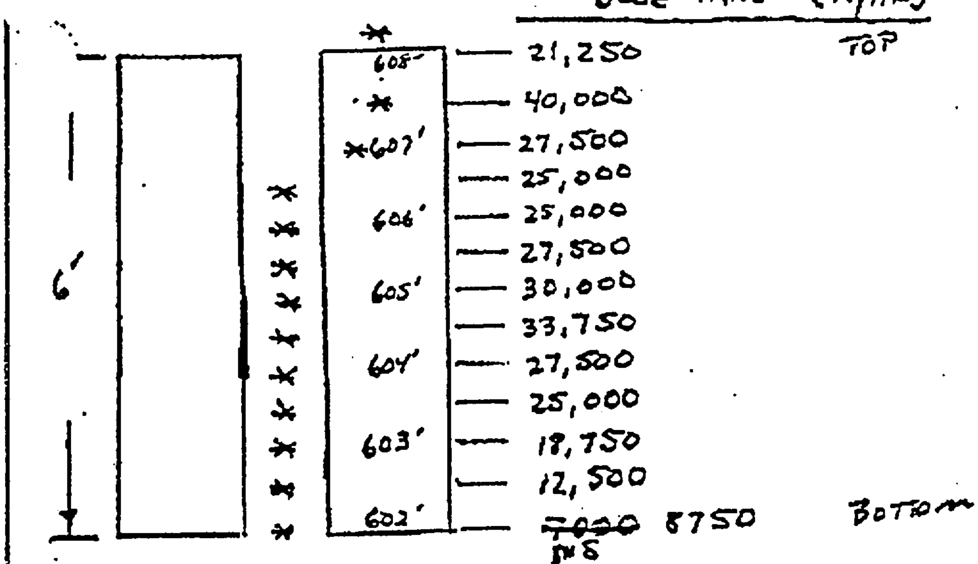
1. N/R dpm/100cm²	5. N/R dpm/100cm²	9. N/R dpm/100cm²	13. N/R dpm/100cm²	17. N/R dpm/100cm²
2. ↓ dpm/100cm²	6. ↓ dpm/100cm²	10. ↓ dpm/100cm²	14. ↓ dpm/100cm²	18. ↓ dpm/100cm²
3. ↓ dpm/100cm²	7. ↓ dpm/100cm²	11. ↓ dpm/100cm²	15. ↓ dpm/100cm²	19. ↓ dpm/100cm²
4. ↓ dpm/100cm²	8. ↓ dpm/100cm²	12. ↓ dpm/100cm²	16. ↓ dpm/100cm²	20. ↓ dpm/100cm²

Comments: Axial survey of nuetron windows in 6" vertical increments using 6" standoff (12" dia). All readings have corrected to Co-60 using a corr factor of 1.25 as per Palisades Procedure HP 9.72

Map Key: # : General Area Dose Rate C/GA: Contact and General Area Dose Rate ⊙: Smear Location --x--x-: Radiological Boundary LDWA: Low Dose Waiting Area *: Hot Spot

* = Survey location using 6" standoff (12" dia)

DOSE RATES (R/HR)



- surveys from 602' → 606' 6" were taken with standoff in contact with both nuetron windows
- Due to obstruction of handles at 607' elevation the standoff was only in contact with southernmost window.

Unposted Hot Spots May be Present

Op's Rounds
- ☐ Gloves for Reach ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

RP Surveys
- ☐ Gloves for Reach ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

Tours/Inspections
- ☐ Gloves for Reach ins SEE RWP
- ☐ Class A
- ☐ Class B
- ☐ Class C

Work Entries
- ☐ Class A SEE RWP
- ☐ Class B
- ☐ Class C

Dose Rate Meter SN: PR-2 #459	Cal Due: 11-3-99	Air Sampler: N/R	Cal Due: N/R
Counting Inst. SN: N/R	Cal Due: N/R	Inst. Type SN: N/R	Cal Due: N/R
Performed By: M. Stevens, K. Tuite	Date: 12/16/98	Reviewed By:	Date:

DOSITEC PR-2
CERTIFICATE OF CALIBRATION - MEDIUM RANGE

Consumers Instrument Number #454

Gamma Calibration Source Number 77CS/5/18

Mechanical Inspection ☑ SAT

Battery Test ☑ SAT ☐ Changed

Scale	Calibration Point	"As Found"	Final	Limits
x1	1 R/hr	N/A	1.0	0.9 - 1.1 R/hr
	2.5 R/hr	>5	2.5 R/hr	N/A
	4 R/hr	N/A	3.7	3.6 - 4.4 R/hr
x10	10 R/hr	N/A	10	9 - 11 R/hr
	25 R/hr	>50	25 R/hr	N/A
	40 R/hr	N/A	38	36 - 44 R/hr
x100	100 R/hr	N/A	100	90 - 110 R/hr
	250 R/hr	>500	250 R/hr	N/A
	400 R/hr	N/A	370	360 - 440 R/hr

Comments: Due to failure of Probe #455 remated Probe #454 with Instrument Box #454

#454

DOSITEC PR-2
CERTIFICATE OF CALIBRATION : MEDIUM RANGE

Reason for Calibration

- ☐ Calibration Due
- ☐ Calibration Check
- ☐ Initial Calibration
- ☑ Returned From Repair
- ☐ Other ________

Results

- ☑ Calibration Acceptable
- ☐ Calibration Not Acceptable

Labels

- ☑ Calibration Label Affixed
- ☑ Dedicated Probe Label Affixed
- ☑ Correction factor 1.25 affixed
- ☑ Limited Use Label Affixed (If needed)

Disposition of Instrument

- ☑ Instrument returned to service

Records

- ☑ Instrument status updated
- ☑ Certificate of calibration submitted for review

Calibration Date: 5-13-98

Calibration Due: 5-13-99

Calibrated By: KT Dickanson

Reviewed By: [signature]
HPPI Specialist

#469

DOSITEC PR-2
CERTIFICATE OF CALIBRATION - HIGH RANGE

Reason for Calibration

- ☑ Calibration Due
- ☐ Calibration Check
- ☐ Initial Calibration
- ☐ Returned From Repair
- ☐ Other ________

Results

- ☑ Calibration Acceptable
- ☐ Calibration Not Acceptable

Labels

- ☑ Calibration Label Affixed
- ☑ Dedicated Probe Label Affixed
- ☑ Correction factor 1.25 affixed
- ☐ Limited Use Label Affixed (if needed)

Disposition of Instrument

- ☑ Instrument returned to service

Records

- ☑ Instrument status updated
- ☑ Certificate of calibration submitted for review

Calibration Date: 11/3/98

Calibration Due: 11/3/99

Calibrated By: [signature]

Reviewed By: [signature] / RB 11/4/98
HPPI Specialist

IE

DOSITEC PR-2
CERTIFICATE OF CALIBRATION - HIGH RANGE

Consumers Instrument Number 469

Gamma Calibration Source Number 87CS.S.1C

Mechanical Inspection ☒ SAT

Battery Test ☒ SAT ☐ Changed

Scale	Calibration Point	"As Found"	Final	Limits
x1	100 R/hr	N/A	100	90 - 110 R/hr
	250 R/hr	250	250 R/hr	N/A
	400 R/hr	N/A	400	380 - 440 R/hr
x10	1000 R/hr	N/A	1000	900 - 1100 R/hr
	1751 R/hr	1751 R/hr	1751 R/hr	N/A

Comments: x100 scale is not calibrated

x10 scale is a two point calibration - the as found point must be chosen from the decay curves.

DOSITECH PR-2
DAILY INSTRUMENT OPERATIONAL / FUNCTIONAL CHECK - MEDIUM RANGE

Consumers Instrument Number 454 w/ probe # 454

Gamma Source Number 9116

Mechanical Inspection SAT ✓ UNSAT____

If unsat - explain: ____

Battery Check SAT____ UNSAT____

SCALE	SOURCE	EXPECTED READING	ACTUAL READING	ACCEPTANCE RANGE (+OR-20%)	DATE/TIME	TECH INITIALS
x10	30 Ci' Cs_{137}	25 R/HR	27 R/HR	20 - 30 R/HR	12/15/98 0705	[illegible]
x10	30 Ci' Cs_{137}	25 R/HR	27 R/HR	20 - 30 R/HR	12/16/98 0705	[illegible]
					/	
					/	
					/	
					/	
					/	

Comments : ____

DOSITECH PR-2
DAILY INSTRUMENT OPERATIONAL / FUNCTIONAL CHECK - HIGH RANGE

Consumers Instrument Number 469 w/ Probe #469

Gamma Source Number 9116

Mechanical Inspection SAT ✓ UNSAT____

If unsat - explain: ____________________

Battery Check SAT ✓ UNSAT____

SCALE	SOURCE	EXPECTED READING	ACTUAL READING	ACCEPTANCE RANGE (+OR- 20%)	DATE/TIME	TECH INITIALS
x1	30 Ci Cs-137	100 R/HR	110	80-120 R/HR	12/5/98 / 0700	JA
x1	30 Ci Cs-137	100 R/HR	110	80-120 R/HR	12/6/98 / 0700	JA
					/	
					/	
					/	
					/	
					/	

Comments: ____________________

APPENDIX B

TELEDYNE BROWN ENGINEERING

ENVIRONMENTAL SERVICES

SUMMARY OF ISOTOPIC ANALYSIS

MAR 1

TELEDYNE BROWN ENGINEERING Environmental Services

REPORT OF ANALYSIS

Mar 09 1999, 07:45 pm

LOGIN # L4182

PAGE: 1

Address:	Work Order #:	Cust. P.O. #:	Date Received	Delivery Date	
DAVE PARISH CONSUMERS ENERGY COMPANY BIG ROCK POINT NUCLEAR PLANT 10269 US-31 NORTH CHARLEVOIX MI 49720	L4182	C0028359 Release #:	12/30/98	01/31/99	Project Manager: C.STARR

Matrix - Solids

Teledyne Sample #	Customer's Identification	Sta. #	Collection Dates Start Date/Time	Collection Dates Stop Date/Time	Matrix/ Nuclide	Activity	Units	Count Date	Volume Procedure #	Units	Lab Comment
					Solids						
L4182-1 TII-96261	COUPON METAL SHAVING		12/01/98 00:00		C-14	L.T. 2. E-04	uCi/g Dry	03/06/99			32
					FE-55	1.4 +-0.2 E-02		02/08/99			32
					GR-A	L.T. 1. E-04		02/13/99			32
					I-129	4.8 +-1.6 E-04		03/04/99			32
					NI-59	L.T. 1. E-03		01/30/99			32
					NI-63	3.8 +-0.2 E-03		02/09/99			32
					SR-89	L.T. 5. E-04		02/07/99			32
					SR-90	L.T. 1. E-04		02/07/99			32
					TC-99	1.9 +-0.8 E-04		02/20/99			32
					BE-7	L.T. 8. E-03		01/21/99			42
					K-40	L.T. 6. E-03		01/21/99			42
					CR-51	L.T. 9. E-03		01/21/99			42
					MN-54	4.48+-0.45E-02		01/21/99			42
					CO-58	L.T. 1. E-03		01/21/99			42
					FE-59	L.T. 5. E-03		01/21/99			42
					CO-60	9.26+-0.93E-01		01/21/99			42
					ZN-65	1.81+-0.25E-02		01/21/99			42
					NB-94	L.T. 1. E-03		01/21/99			42
					NB-95	L.T. 1. E-03		01/21/99			42
					ZR-95	L.T. 2. E-03		01/21/99			42
					MO-99	L.T. 5. E 01		01/21/99			42
					RU-103	L.T. 1. E-03		01/21/99			42
					RU-106	L.T. 6. E-03		01/21/99			42
					AG-110M	L.T. 2. E-03		01/21/99			42
					SB-124	L.T. 2. E-03		01/21/99			42
					SB-125	L.T. 1. E-03		01/21/99			42
					I-131	L.T. 3. E-02		01/21/99			42
					CS-134	L.T. 9. E-04		01/21/99			42
					CS-137	L.T. 7. E-04		01/21/99			42
					BA-140	L.T. 2. E-02		01/21/99			42
					LA-140	L.T. 9. E-03		01/21/99			42
					CE-141	L.T. 8. E-04		01/21/99			42
					CE-144	L.T. 1. E-03		01/21/99			42
					RA-226	L.T. 4. E-03		01/21/99			42

TELEDYNE BROWN ENGINEERING Environmental Services

REPORT OF ANALYSIS

Mar 09 1999, 07:45 pm

LOGIN # L4182

Address:	Work Order #:	Cust. P.O. #:	Date Received	Delivery Date	PAGE: 2
DAVE PARISH CONSUMERS ENERGY COMPANY BIG ROCK POINT NUCLEAR PLANT 10269 US-31 NORTH CHARLEVOIX MI 49720	L4182	C0028359 Release #:	12/30/98	01/31/99	Project Manager: C.STARR

Teledyne Sample #	Customer's Identification	Sta. #	Collection Dates Start Date/Time	Collection Dates Stop Date/Time	Matrix/ Nuclide	Activity	Units	Count Date	Volume	Procedure #	Units	Lab	Comment
Continued													
					TH-228	L.T. 5. E-04	uCi/g Dry	01/21/99				42	
					NP-237	L.T. 7. E-04		01/21/99				42	
					H-3	L.T. 2. E-03						52	
					AM-241	5.1 +-1.3 E-05		02/24/99				62	
					CM-242	1.7 +-0.8 E-05		02/24/99				62	
					CM-243/244	L.T. 3. E-06		02/24/99				62	
					NP-237	L.T. 6. E-06		02/24/99				62	
					PU-238	1.2 +-0.7 E-05		02/24/99				62	
					PU-239/240	2.6 +-1.0 E-05		02/24/99				62	
					PU-241	L.T. 7. E-04		02/24/99				62	
					PU-242	L.T. 4. E-06		02/24/99				62	
					U-233/234	L.T. 3. E-06		02/24/99				62	
					U-235	L.T. 3. E-06		02/24/99				62	
					U-238	L.T. 3. E-06		02/24/99				62	

Approved By: J. Duenther

Last Page of Report

Lab Key: 22 - Gas Lab; 32 - Radiochemistry Lab; 42 - GS(Li) Gamma Spec Lab; 52 - Tritium Lab; 62 - Alpha Spec Lab; 12 - Environmental TLD; 72 - Consulting;

Copy: 1 of 1

APPENDIX 5-3

ISSUE SUMMARY
Form SOP-0402-03, Revision 2

DESIGN CONTROL SUMMARY

CLIENT:	**BNFL**	UNIT NO.: **N/A**	QA SERIAL NO.
PROJECT NAME:	**Big Rock Point Major Component Removal**		
PROJECT NO.:	**10902-010**	☐ NUCLEAR SAFETY-RELATED	
CALC. NO.:	**N-10902-010-001**	☐ NOT NUCLEAR SAFETY-RELATED	
TITLE:	**RV External Measurements Evaluation**	☒ IMPORTANT TO SAFETY- CATEGORY A	
EQUIPMENT NO.:	**N/A**		

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

Initial Issue: 11 pages in calculation body & 10 pages total in Attachments

INPUTS/ ASSUMPTIONS
☒ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:	Detailed	REV. 0	
STATUS:	Approved	DATE FOR REV.:	2/21/01
PREPARER	R. Kahn	DATE:	2/21/01
REVIEWER *	J. M. Rich	DATE:	2/21/01
APPROVER	W. J. Johnson	DATE:	2/21/01

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

This revision completely supercedes revision 0. And includes all
Pages ie., pages 1 thru 15 for calculation body and attachment pages A1 thru A7, B1, B2, & C1.

INPUTS/ASSUMPTIONS
☒ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:	Detailed	REV. 1	
STATUS:	Approved	DATE FOR REV.:	6-4-01
PREPARER	A. G. Klazura *Anthony A. Klazura*	DATE:	5/25/01
REVIEWER*	R. Kahn *Roger Kahn*	DATE:	5/25/01
APPROVER	W. J. Johnson *[signature]*	DATE:	6/4/01

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

INPUTS/ASSUMPTIONS
☐ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:		REV.	
STATUS:		DATE FOR REV.:	
PREPARER		DATE:	
REVIEWER*		DATE:	
APPROVER		DATE:	

* The reviewer's signature indicates compliance with S&L procedure SOP-0402 and the verification of, as a minimum, the following items: correctness of mathematics for manual calculations, appropriateness of input data, appropriateness of assumptions, and appropriateness of the calculation method.

NOTE: PRINT AND SIGN IN THE SIGNATURE AREAS

Rev. Date: 01-29-1999
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1.0 Purpose and Scope

1.1 Purpose

The purposes of this calculation are to; (a) perform an assessment of the reported radionuclide content of the reactor vessel (RV) wall specimen, and (b) compare the results of the specimen analysis to the data and model used to design shielding and support shipment of the Big Rock Point Reactor Vessel Transport System (RVTS).

In addition, dose rates from radiological surveys in the vicinity of the RV are compared to calculated dose rates that use the design model and source term. This comparison provides an indication of the reasonableness and the degree of conservatism in RVTS design and dose rate calculations. In particular, it verifies conservatism to allow for possible "hot spots", e.g. increase in activity of some RV areas due to the effect of the Neutron Windows on surrounding areas even though the Neutron Windows have been removed and thus will not be part of the RVTS.

Measured radionuclide activity is presented in Reference 2.4. Results of the radiological surveys are provided in Reference 2.7. The source composition used in the analytical model is based on Reference 2.3. The calculated dose rates are obtained using the Reference 2.6 computer code and the computer model is based on that used in Reference 2.1.

Revision 1 to the calculation added the following:

- Provides details concerning the development of reactor component source term activities provided in Section 3.1 of this calculation,
- added Section 3.7 which provides a comparison between sampled and calculated wetted surface contamination,
- added a table to compare nuclide specific measured activation in the reactor vessel wall and vessel insulation to modeled activation , and
- removed margin from the source term used in the computer model to calculate contact dose rates to the reactor vessel.

This calculation is intended to demonstrate that the analytical model used in radiological evaluations of the reactor vessel and used to develop shielding for vessel transport is adequate and conservative by demonstrating that measured dose rates and source term inventories are bounded by calculated values.

2.0 References

2.1 "Transport Package Shielding Design", Calculation No. N-10525-020-0002, Big Rock Point Major Component Removal, Revision 1.

2.2 "Big Rock Point Reactor Vessel and Internals Characterization and Classification", Report WMG-9902, Revision 1, June 1999, WMG Project 8057.

2.3 "Reactor Vessel Transport System Radiation Source Term," Calculation No. N-10525-020-004, Revision 1.

2.4 "Final Laboratory Analysis Report, 10CFR61 Analysis for Big Rock Point," letter from William R. Stagg, BWXT Services, Inc. to Bill Gray, Framatome Technologies, Inc., dated August 16, 2000.

2.5 Drawings: Consumers Power, Big Rock Point.

2.5.1 Drawing 104R175, "Reactor - Vessel," Revision 3
2.5.2 Drawing 107C3539, "Windows," Revision 0
2.5.3 Drawing 141F797, "Thermal Shield," Revision 5
2.5.4 Drawing 197E853, "Vessel & Core Arrangement," Revision 2

2.6 Program ISOSHLD-PC, Program 03.7.310-1.0/O, QAD options, June 1, 1999, Sargent Lundy[LLC].

2.7 "Big Rock Point Radiological Survey and Status Sheets", Facsimile transmittal from D. Baldwin to J. Johnson, January 23, 2001.

2.8 "Revised Characterization and Classification Results; WMG Project 8057," March 10, 2000, WMG inc.

3.0 Design Input

3.1 Reactor Component Activities [Reference 2.1, Attachment J]

Radionuclide distribution and radionuclide inventories for components in the active core region are presented in Table 3-1. The inventories in the table were used in the transport package shielding evaluation [Reference 2.1], and have already been decayed to shipment date of September 1, 2002. The activity for each component has been homogenized over the portion of the component in the active core region. Component activities presented in Table 3-1 represent the total activity for the identified component and are obtained from the WMG characterization [Reference 2.8]. The component activities were increased by 15% to obtain the adjusted activities for the thermal shield, the RV insulation, and the RV wall. These adjusted activities are used in the shielding analysis and provide additional conservatism. The component activity in the thermal shield retainer was divided by 6 to obtain the activity in each of 6 retainers and was increased by 15%. The reactor vessel cladding source term was combined with the reactor vessel wall source term in the transport package shielding evaluation. To account for the attenuation of the cladding source term by the reactor vessel wall, the cladding source term was divided by 30. This reduced source term was then increased by 15%. The radionuclide source fractions listed in Table 3-1 are from Table 3-2 of Reference 2.1. The source fractions used to obtain nuclide inventories for the reactor vessel wall are different from the fractions provided in Table 3-1 and are from Table 6.3.2.2 of Reference 2.3. The bases for source term adjustments used in the transport package shielding evaluation are provided in Reference 2.1.

Table 3-1. Active Core Region Activities Used for Dose Rate & Shielding Calculations [Ref. 2.1]

			Activity (Curies)					
Nuclide	QAD Nuclide ID #	Source Fraction	Thermal Shield	RV Cladding	RV Wall**	Total (RV Wall & Clad)	RV Insulation	Thermal Shield Retainer
Component Activity (Curies)			7.40E+03	2.45E+02	1.11E+03		5.81E+01	1.14E+02
Adjusted (Modeled) Activity (Curies)*			8.51E+03	9.39E+00			6.68E+01	2.19E+01
H-3	451	3.26E-04	2.77E+00	3.06E-03	4.16E-01	4.19E-01	2.18E-02	7.12E-03
C-14	452	1.08E-04	2.31E+02	2.54E-01	3.46E+01	3.49E+01	1.81E+00	5.92E-01
Sb-125	269	2.01E-07	1.71E-03	1.89E-06	2.56E-04	2.58E-04	1.34E-05	4.39E-06
Mn-54	473	1.07E-03	9.11E+00	1.00E-02	1.37E+00	1.38E+00	7.15E-02	2.34E-02
Eu-152	408	1.26E-05	1.07E-01	1.18E-04	1.61E-02	1.62E-02	8.42E-04	2.75E-04
Fe-55	474	3.47E-01	2.96E+03	3.27E+00	9.09E+02	9.13E+02	2.32E+01	7.60E+00
Co-60	481	5.72E-01	4.87E+03	5.37E+00	2.66E+02	2.71E+02	3.82E+01	1.25E+01
Ni-59	None	4.74E-04	4.03E+00	4.45E-03	6.06E-01	6.11E-01	3.17E-02	1.04E-02
Ni-63	None	7.81E-02	6.65E+02	7.33E-01	9.98E+01	1.01E+02	5.22E+00	1.71E+00
Nb-94	111	1.64E-06	1.40E-02	1.54E-05	2.09E-03	2.11E-03	1.10E-04	3.58E-05
Tc-99	141	3.51E-07	2.99E-03	3.30E-06	4.49E-04	4.52E-04	2.35E-05	7.67E-06

* Adjusted activity is 15% greater than component activity as indicated in Section 6.1. of Ref 2.1

** Nuclide distribution for the RV wall is provided in Table 3-1.1 of Reference 2.1

3.2 Active Core Region Components

The active core region components are those located along the elevation occupied by the reactor fuel. The characteristics are taken from References 2.1 and 2.2.

Table 3-2. Active Core Region Irradiated Component Characteristics [Ref. 2.1]

Component	Inner Radius	Outer Radius
Thermal Shield (TS)	50.00 in.	51.50 in.
TS Retainer	(rectangular brackets)	-
Seal Weights	51.50 in.	52.50 in.
RV Wall	53.22 in.	58.47 in.
RV Insulation	58.47 in.	61.47 in.

The reactor vessel wall is 5.25 in. thick carbon steel. [Reference 2.2].

The active fuel region is 70" in length. The bottom of the active fuel region is 8'-1.75" above the base line [Drawing 197E853, Reference 2.5]. The top of the reactor vessel flange is 282" above the base line [Drawing 104R175, Reference 2.5].

The reactor vessel insulation is 3" thick. It is composed of stainless steel plate with folded stainless steel foil between the plates. The insulation extends 3" from the reactor vessel surface [Section 3.7 of Reference 2.1].

3.3 RV Component Weights and Volumes

Specific activities and densities for the various RV components will be determined from physical data included in Table 3-8 of Reference 2.1. The data is reproduced in Table 3-3 below.

Table 3-3. RV Component Physical Characteristics [Ref. 2.1]

Component	Weight	Volume
Thermal Shield (TS)	1.30 E+4 lbs	2.59 E+1 ft^3
TS Retainer	5.47 E+2 lbs	1.09 E+0 ft^3
Seal Weights	5.08 E+3 lbs	1.02 E+1 ft^3
RV Wall	2.03 E+5 lbs	4.14 E+2 ft^3
RV Insulation	5.18 E+3 lbs	2.59 E+2 ft^3

3.4 Measured Data

Measured radionuclide data refers to isotopic analysis of specimens provided in Reference 2.4. Some sample results are presented in Table 6.2 of this calculation. Specimens were taken from two core bores [Reference 2.7]: at the North direction from the RV wall (referred to as "North" further in the caculation) and at the West direction from the RV wall ("West"). The West core bore is 2'-10" north of center axis of the RV. Both core bores are at elevation 605' [Reference 2.7], which is near the midplane of the active fuel.

3.5 Survey Results

Results of the radiological survey are provided in Reference 2.7. The dose rates were measured at the peak dose rate locations on the north side of the RV where fuel is closest to the RV wall and on the west side where fuel is closest to the RV wall and near the neutron window location. Dose rate measurement results at contact with RV walls are presented in Table 6.4 of this calculation. Some of the measurements were performed with RV insulation removed.

3.6 QAD Model

QAD dose rate calculations were performed using the same methodology and material data as in Reference 2.1 with some exceptions noted in appropriate sections of this calculation.

3.7 Surface Contamination

A comparison between surface contamination information provided in the WMG report [Reference 2.2] and the calculated surface contamination that was used in radiological evaluations is presented in Table 3-4. Radionuclide specific surface contamination values (μCi/gm) listed in Appendix B of the WMG report were characterized by Teledyne Brown Engineering Environmental Services based on a 0.52 gram sample taken from the internal surface of the core spray nozzle. The surface contamination values given in the WMG report were decayed to the shipment date of 9/1/2002 in order to match the design calculation source terms. The calculated surface contamination values (μCi/gm) were obtained by multiplying the nuclide specific total surface contamination value (Ci) from Table 6.2.3-1 of Reference 2.3 by 1E+6 μCi/Ci and dividing by the wetted surface area of 302,000 square inches and by the factor to convert the sample of a given mass and cross sectional area from μCi/gm to μCi/in^2. The wetted surface area and the conversion factor are presented in Section 6.2.1 of Reference 2.3.

Table 3-4. Surface Contamination Comparison

Nuclide	Measured Contamination (µCi/gm) Decayed to 9/1/2002	Calculated Contamination (µCi/gm) At 9/1/2002
C-14	---	2.35E-4
Fe-55	5.67E-3	6.22E-2
I-129	4.80E-4	4.78E-4
Ni-63	3.71E-3	4.44E-2
Sr-90	---	1.89E-3
Tc-99	1.90E-4	1.90E-4
Mn-54	2.40E-3	2.57E-3
Co-60	5.76E-1	7.78E-1
Zn-65	4.26E-4	4.25E-4
Sb-125	---	3.41E-3
Ce-144	---	9.47E-5
H-3	---	1.66E-4
Am-241	5.07E-5	2.75E-4
Cm-242	8.69E-8	1.18E-7
Cm-243	---	7.35E-5
Cm-244	---	6.97E-5
Pu-238	1.17E-5	1.82E-4
Pu-239/240	2.60E-5	2.17E-4
Pu-241	---	1.03E-2
Pu-242	---	9.60E-7
TOTAL	5.89E-1	9.06E-1

4.0 Assumptions

4.1 It is assumed that Co-60 in the RV components at the active core region dominates the dose rate (see also Reference 2.1). Therefore, only Co-60 inventories were used to compare measured activity [Reference 2.4] with activities used in the dose rate calculations [Reference 2.1].

4.2 Material compositions and densities were used as in Reference 2.1, but the low density cellular concrete (LDCC) was replaced with water having a density of 1.0 gram/cc in order to reflect the current situation in which the vessel is filled with water.

4.3 As in Reference 2.1, it is assumed that radionuclide activity is homogeneously distributed within the component or component portion under consideration. Most of the activity of a component is located in the active core region of that component.

4.4 For the purposes of this calculation, a Roentgen (R) is assumed to be equal to a rem.

5.0 Methodology and Acceptance Criteria

5.1 Acceptance Criteria

There are no specific acceptance criteria for this calculation.

A qualitative and reasonable agreement between the peak measured data (specific activities and dose rates) and the calculated parameters would be considered as a validation of reasonableness, and an indicator of the degree of conservatism in the dose rate calculations.

5.2 Methodology for QAD Evaluation

The QAD option of the ISOSHLD-PC computer program [Reference 2.6] is used to determine dose rates at contact and 1 meter from the reactor vessel insulation outer surface at the active core midplane. A detailed description of the QAD evaluation methodology and model is provided in, and is identical to, Reference 2.1. The reactor vessel and other component source terms in the active core region are the same as in Reference 2.1 with the exception that the 15% margin added to the RV component inventories in Reference 2.1 was removed for the contact dose rate evaluation.

Dose rates at contact (actually at 1" from the RV insulation surface) were determined in this calculation for comparison with the measured data in Reference 2.7. The Reference 2.1 calculation does not include the contact dose rates. In addition, the 1-meter dose rate was calculated under the same conditions (when LDCC was replaced with water having a density of 1.0 grams/cc) for comparison with the results of Reference 2.1 calculations.

The ISOSHLD-PC computer code (including the QAD option of ISOSHLD-PC) has been qualified under S&L's Quality Assurance program.

Input and Output files for the QAD model are identified in Attachment B to this calculation.

6.0 Calculations

6.1 Comparison of Activity in the Reactor Vessel Wall and Insulation

Measured nuclide specific activities outside the reactor vessel were obtained from Reference 2.4 and are based on sample specimens. Specific activities were determined for a reference date of 5/24/00 in Reference 2.4. They have been decayed to the shipment date of 9/1/2002 to match design calculation source terms. Measured specific activity values from Reference 2.4, decayed to the shipment date of 9/1/2002, are provided in Table 6-1 for the North RV wall, the North RV insulation, and the West RV insulation. Modeled RV wall activation specific activities and modeled RV insulation specific activities at the shipment date of 9/1/2002, are included in Table 6-1 for comparison. The modeled RV wall nuclide activation specific activities presented in Table 6-1 are obtained using equation 1 and the modeled RV insulation activation specific activities are obtained using equation 2. The reactor vessel wall total nuclide inventories input into equation 1 are from Table 3-1.1 of Reference 2.1. The reactor vessel insulation total nuclide inventories input into equation 2 are obtained by multiplying the RV insulation total inventory value (i.e., 58.1 curies from Table 3-1 of Reference 2.1) by the nuclide fractions obtained from Table 6.3.2-1 of Reference 2.3.

Table 6-1. Activation Comparison

Nuclide	Measured North RV Wall Activation Specific Activity (μCi/gm)	Modeled RV Wall Activation Source Specific Activity (μCi/gm)	Measured North Insulation Activation Specific Activity (μCi/gm)	Measured West Insulation Activation Specific Activity (μCi/gm)	Modeled RV Insulation Activation Specific Activity (μCi/gm)
C-14	4.91E-3	7.29E-3	3.01E-3	1.34E-3	1.08E-2
Fe-55	2.11E+1	4.76E+1	7.08E+1	3.55E+1	3.45E+1
Ni-59	5.65E-3	3.17E-2	1.97E-1	1.48E-1	4.72E-2
Ni-63	6.05E-1	5.23E+0	2.05E+1	2.23E+1	7.77E+0
Tc-99	3.51E-4	2.35E-5	---	---	3.50E-5
Mn-54	1.17E-1	7.16E-2	5.72E-2	7.11E-2	1.07E-1
Co-60	5.21E+0	1.39E+1	2.20E+1	5.60E+1	5.70E+1
Zn-65	5.83E-03	---	---	2.98E-1	---
Nb-94	---	1.10E-4	---	---	1.63E-4
Sb-125	---	1.34E-5	---	---	2.00E-5
H-3	5.71E-3	2.18E-2	3.58E-3	2.78E-3	3.24E-2
Pu-239/240	8.61E-7	---	8.54E-7	1.46E-6	---
U-233/234	6.85E-6	---	4.04E-6	2.13E-6	
U-238	4.98E-7	---	9.46E-7	---	---
Eu-152	---	8.43E-4	---	---	1.25E-3
TOTAL	2.70E+1	6.68E+1	1.14E+2	1.14E+2	9.94E+1

An example of the methodologies used to decay specimen specific activities to the shipment date and to obtain the modeled specific activities follows for the Co-60 nuclide. Co-60 was chosen because the dose rate outside the RV is dominated by Co-60 (Assumption 4.1).

The measured Co-60 specific activities of sample specimens from the outside of the RV [Reference 2.4] are presented in Table 6-2. The measured results were decayed to September 1, 2002.

Table 6-2. Laboratory Analysis Report Data Decayed to September 1, 2002

Nuclide	Sample Description	Reported Quantity (μCi/gm)	Laboratory Reference Date	Half Life (year)	Shipment Reference Date 9/1/2002	Days to 9/1/2002	Activity On 9/1/2002 (μCi/gm)
Co60	North RX Vessel	7.02E+00	5/24/00	5.27E+00	9/1/02	830.00	5.21E+00
Co60	North Outer Insul	2.97E+01	5/24/00	5.27E+00	9/1/02	830.00	2.20E+01
Co60	North Wall Core Bore 34"	9.65E-06	5/24/00	5.27E+00	9/1/02	830.00	7.16E-06
Co60	West Miror Insul	7.55E+01	5/24/00	5.27E+00	9/1/02	830.00	5.60E+01
Co60	West Wall Core Bore 34"	2.15E-05	5/24/00	5.27E+00	9/1/02	830.00	1.59E-05

Reference 2.4 data does not contain a specimen from the West RV wall, which may present a special interest due to the fact that the highest measured dose rate was located in the West core bore [Reference 2.7 and Table 6-3]. The specific activity at the West RV wall was estimated from the ratio of similar locations at the North and West core bores presented in Table 6-2, and by applying this ratio to the North RV wall measured data. The data for calculating the sample activity ratios is from Table 6-2. The data for calculating the dose rate ratios is from Table 6-4. For example:

$$WestRVWall = NorthRVWall\left(\frac{WestInsulActivity}{NorthInsulActivity}\right) = 5.21x\left(\frac{5.6E1}{2.2E1}\right) = 1.33E1(\mu Ci/gm)$$

Table 6-3 Derivation of the West Reactor Vessel C0-60 Activity

Sample Description	Ratio of West/North Sample Activities	Expected West RV Wall Activity on 9/1/02 (μCi/gm)	Survey Point Description	Survey Date	Ratio of West/North Dose Rates
Wall Core Bore 34"	2.23E+00	1.16E+01	RX Vessel Contact	6/7/00	1.74E+00
Insulation	2.54E+00	1.33E+01	RX Vessel Contact	7/28/00	2.81E+00
Average	2.39E+00	1.25E+01	Average		2.28E+00

In order to compare the measured specimen data with the activity used in the dose rate calculation, the latter has to be converted from the total Co-60 activity (μCi) in the active core region of the RV wall and RV insulation into the specific activity (μCi/gm).

Since the active fuel height is 70 inches (Section 3.2), and the RV wall has an inner radius of 53.22 inches and a thickness of 5.25 inches, the mass of the RV wall adjacent to the active fuel is:

$$\frac{\left((58.47\ in)^2-(53.22\ in)^2\right)\times\pi\times 70\ in\times 7.86\ g/cc\times 16.387\ cc/in^3}{454.54\ g/lb}=3.654E+4\ lb$$

The specific activity in this region is, therefore,

$$\frac{2.31E2(Ci)x1E6(\mu Ci/Ci)}{3.654E4\ (lb)x454.54(g/lb)}=13.88\ \mu Ci/g \qquad \text{[Eq 1]}$$

Where 2.31E2 Ci is Co-60 activity in the active core region of the RV wall from Table 3-1.1 of Reference 2.1

For RV insulation, the Co-60 activity of 33.3 Ci in the active fuel region (58.1 curies from Table 3-1 of Reference 2.1 multiplied by the Co-60 nuclide fraction from Table 6.3.2-1 of Reference 2.3) may be transformed into the specific activity by determining the fraction of the insulation's weight in that region. This can be approximated as a fraction of active fuel region height (70") versus the total RV height (282" [Ref. 2.5.4]):

$$\frac{3.33E1(Ci)x1E6(\mu Ci/Ci)x282(in)}{5.18E3(lb)x454.54(g/lb)x70(in)}=5.70E1\ \mu Ci/g \qquad \text{[Eq 2]}$$

The results of this comparison (measured specimen data vs the calculated activity) is summarized in Table 6-6.

6.2 Comparison of Survey Results and Calculated Dose Rates

The survey dose rates at contact with the RV wall [Reference 2.7] are presented in Table 6-4. The measured results were decayed to the shipment date of September 1, 2002.

Table 6-4. Survey Results Adjusted to September 1, 2002

Survey Point Location	Reported Dose Rate [Ref. 2.7] (R/hr)	Date Measured	Co-60 Half Life	λ (1/day)	Reference date 9/1/2002	Days to 9/1/2002	Dose Rate on 9/1/2002 (R/hr)
North RX Vessel Contact	2.30E+01	6/7/00	5.271 y	3.600E-04	9/1/02	816.00	1.71E+01
West RX Vessel Contact	4.00E+01	6/7/00	5.271 y	3.600E-04	9/1/02	816.00	2.98E+01
North RX Vessel Contact	1.60E+01	7/28/00	5.271 y	3.600E-04	9/1/02	765.00	1.21E+01
West RX Vessel Contact	4.50E+01	7/28/00	5.271 y	3.600E-04	9/1/02	765.00	3.42E+01
North RX Vessel Contact	1.80E+01	1/17/01	5.271 y	3.600E-04	9/1/02	592.00	1.45E+01

Dose rates calculated via use of the QAD computer code are presented in Table 6-5. Dose rates are presented for the following situations:

- unshielded dose rates at contact, i.e., 1" from the surface of the side of the reactor vessel at the active core region midplane;
- unshielded dose rates at 1 meter from the side of the reactor vessel at the active core region midplane.

The 1-meter dose rate was calculated for comparison with the results of Reference 2.1 calculations to verify the effect of LDCC replacement with water (density 1.0 g/cc). The Reference 2.1 value for the same location and source is 19.4 Rem/hr (with LDCC), which is very close to 19.01 Rem/hr (with water) of this calculation. This is an indication that replacement of LDCC with water does not change other calculated dose rates and conclusions. The major sources are the RV wall and the insulation.

In summary; the contact dose rates are 1 inch from the outer insulation surface at the active core midplane, and the 1 meter dose rates are 1 meter from the outer insulation surface at the active core midplane. The QAD model is the same as that in Reference 2.1 except that the LDCC has been replaced by water at 1 gm/cc. Also, the 15% conservatism incorporated into the source term was removed from the QAD results associated with the contact dose rates.

Table 6-5. Unshielded Dose Rates

Radiation Source	QAD File Name	Contact (Rem/hr)*	1 Meter (Rem/hr)
RV Wall	RVCONAC.dat	21.25	8.5
RV Insulation	INCONCL.dat	38.97	7.62
RV Thermal Shield	TSCONAC.dat	3.03	1.79
Thermal Shield Seal Weights	SWCONAC.dat	2.03	1.1
Total		65.29	19.01

* QAD calculated contact dose rates were divided by 1.15 to remove the 15% conservatism factor incorporated into the component source term inventories used in the QAD model.

QAD file names are identified in Attachment B to this calculation. QAD input and output files are included on the CD which accompanies this calculation (Attachment C).

Table 6-6. Comparison of Calculated Results with Dose Survey and Sample Analysis

Source of Data	Parameters Being Compared (decayed to 9/1/02)		
	West RV Wall Contact Dose Rate (R/hr)	West RV Vessel Activity (μCi/gm) Co-60	West Insulation Activity (μCi/gm) Co-60
CALCULATED	6.53E+01	1.39E+01	5.70E+01
MAXIMUM SURVEY & ANALYSIS	3.42E+01	1.33E+01	5.60E+01
AVERAGE SURVEY & ANALYSIS	3.20E+01	1.25E+01	N/A
Ratio: Calculated/ Max Measured	1.91E+00	1.05E+00	1.02E+00
Ratio: Calculated/ AVG Measured	2.04E+00	1.11E+00	N/A

Note that in all cases the activity used in the calculation is larger than the highest activity in the specimen. The calculated dose rates are even more conservative when compared to the survey results (by about a factor of 2).

Elevated Co-60 specific activity and dose rates in the West direction are due, at least partially, to higher activation in the east or west portions of the RV near the neutron windows.

The actual azimuth of the windows' centerline is at 277.5° [Reference 2.5.3]. The West core bore, which is 2'-10" north of the West centerline is at about 299° [This is calculated from ARCTAN(2'-10" / 61.47") = 29°, where 61.47" is an outer radius of RV insulation, and 270° + 29° = 299°]. Therefore, the West core bore is in the general area of potential effects of neutron windows on RV component activation.

7.0 Summary and Conclusions

The results indicate that the activity used to design the RVTS is larger than the highest activity in the specimen. The calculated dose rates are even more conservative when compared to the survey results (by about a factor of 2).

This degree of conservatism in dose rates is, in part, due to conservatism inherent in the point- kernel approach utilized by the ISOSHLD/QAD model and due to conservative axial distribution of source terms, etc. Additional conservatism is incorporated into the shielding evaluation for the reactor vessel transport system (e.g., 15% source term margin).

Measured activity and dose rates are representative of maximum values, although the available data is not sufficient to evaluate effects of various factors such as the neutron windows, distance from the active core edges to the RV walls, etc.

Measured dose rates are influenced by the surrounding active components, like the activated concrete. The measured dose rates are, therefore, expected to be significantly higher than the dose rates that are due solely to the RV. This further increases the conservatism in the calculated dose rates.

The analytical model (and source terms used in the model) have been shown to bound the measured peak source term activities and dose rates associated with the RVAI. Therefore, the analytical model and methodology used to generate the source terms and reactor transport vessel shielding design is conservative with sufficient margin to accommodate peak dose rates and activities.

The 2 meter dose rate calculated in Reference 2.1 is 4.3 mrem/hr along the active core midplane. This is a factor of two lower than the required limit of 10 mrem/hr. When combined with the factor of two of conservatism based on comparison of measured and calculated results, the total factor of conservatism is approximately four.

8.0 Attachments

A. "Big Rock Point Radiological Survey and Status Sheets", Reference 2.7, 7pages;

B. Attachment B: "ISOSHLD-PC/QAD Input/Output Files", 2 pages;

C. Attachment C: CD which contains ISOSHLD-PC and QAD computer input and output files, 1 page.

FINAL

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FACSIMILE TRANSMITTAL SHEET

TO: JOE JOHNSON	FROM: DAVE BALDWIN
COMPANY: S&L	DATE: 01/23/2001
FAX NUMBER: (312) 269-2028	TOTAL NO. OF PAGES INCLUDING COVER: 7
PHONE NUMBER: (312) 269-6649	SENDER'S REFERENCE NUMBER:
RE: RX VESSEL DOSE RATES	YOUR REFERENCE NUMBER:

☐ URGENT ■ FOR REVIEW ☐ PLEASE COMMENT ☐ PLEASE REPLY ☐ PLEASE RECYCLE

NOTES/COMMENTS:
See comments sheet
I hope this is what is needed
Thanks,
Dave B
(231) 547-8117

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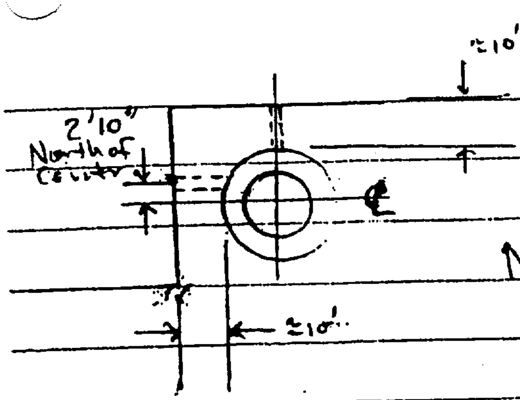
N



Note that length of West concrete core ≈ 4" longer than North Concrete core, and end of core showed sharp angle of former wall.

Survey #1 Original Survey of West Core Bore
5/22/00 Note that it is incorrect - it identifies North Core Bor

#2 Shows both locations after Neutron Windows
6/7/00 removed

#3 Shows both again - to reverify North vs. West
7/28/00 locations

#4 Reverify North Core Bore
1/17/01

#5 Original sketch Dose Rates taken after each stage of core bore - prior to neutron Windows being removed - West Wall

RADIOLOGICAL SURVEY AND STATUS SHEET

UFI: 24*11*01

Survey Log # 2000636

Date/Time: 5-23-00 / ... Technician: M Klink Location: Rm 441	
High Dose Rate: 70,000 mRem/hr(γ) n/a mRem/hr(β)	General Dose Rate: 15 mRem/hr(γ) n/a mRem/hr(β)
High Contamination: 30,000 dpm/100cm²	General Contamination: 10,000 dpm/100cm²
Airborne Concentration: n/a µCi/cc	Spectrum No: n/a

Area Posting: ☒ Contamination Area (CA) ☐ High Radiation Area (HRA) ☐ High Contamination Area (HCA) ☐ Radiation Area (RA) ☐ Locked High Radiation Area (LHRA) ☐ Other (specify) ______ ☐ Airborne Area (ABN)

Special Instructions: ☒ Notify Rad Pro PRIOR to any "Work Entry" ☐ Air Sampling Required for Entry ☐ Notify Monitoring Station When Entering and Exiting ☐ Other (Specify) ______ ☐ Survey Meter Required in HRA's ☐ High Rad Door Watch Required

1. ___ dpm/100cm²	5. ___ dpm/100cm²	9. ___ dpm/100cm²	13. ___ dpm/100cm²	17. ___ dpm/100cm²
2. ___ dpm/100cm²	6. ___ dpm/100cm²	10. ___ dpm/100cm² N	14. ___ dpm/100cm²	18. ___ dpm/100cm²
3. ___ dpm/100cm²	7. ___ dpm/100cm²	11. ___ dpm/100cm² A	15. ___ dpm/100cm²	19. ___ dpm/100cm²
4. ___ dpm/100cm²	8. ___ dpm/100cm²	12. ___ dpm/100cm²	16. ___ dpm/100cm²	20. ___ dpm/100cm²

Comments: Completion of Core Boring in Rm 441

Map Key. #: General Area Dose Rate C/GA: Contact and General Area Dose Rate ①: Smear Location --x--x--: Radiological Boundary LDWA: Low Dose Waiting Area *: Hot Spot a: Air Sample Location

Plane of Hole 15 mR/hr 30 mR/hr @ approximately 4' u/s Hole. 70 R/hr contact on RZ Vessel Liner 30,000 DPM 100 cm² on smears

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1

Unposted Hot Spots May be Present

Ops Rounds
☐ Gloves for reach ins
☐ Class A
☐ Class B per
☐ Class C RWP

RP Surveys
☐ Gloves for reach ins
☐ Class A
☐ Class B per
☐ Class C RWP

Tours/Inspections
☐ Gloves for reach ins
☐ Class A
☐ Class B per
☐ Class C RWP

Work Entries
☐ Class A per
☐ Class B RWP
☐ Class C

Dose Rate Meter/SN: Tel 29667	Cal Due: 7-17-00	Air Sampler: n/a	Cal Due: n/a
Counting Inst./SN: 600 172 12/03	Cal Due: 8-8-00	Inst. Type/SN: n/a	Cal Due: n/a
Performed By: M Klink	Date: 5-23-00	Reviewed By:	Date:

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RADIOLOGICAL SURVEY AND STATUS SHEET

UFI: 24*11*01

Survey Log # 2000-711

Date/Time: 6-7-00 0800 Technician: Kurtz	Location: Core Bores on 599'
High Dose Rate: 40000 mRem/hr(γ) N/R mRem/hr(β)	General Dose Rate: N/R mRem/hr(γ) N/R mRem/hr(β)
High Contamination: N/R. dpm/100cm²	General Contamination: N/R. dpm/100cm²
Airborne Concentration: µCi/cc	Spectrum No:

Area Posting:
- ☒ Contamination Area (CA)
- ☐ High Radiation Area (HRA)
- ☐ High Contamination Area (HCA)
- ☒ Radiation Area (RA)
- ☐ Locked High Radiation Area (LHRA)
- ☐ Other (specify)
- ☐ Airborne Area (ABN)

Special instructions:
- ☐ Notify Rad Pro PRIOR to any "Work Entry"
- ☐ Air Sampling Required for Entry
- ☐ Notify Monitoring Station When Entering and Exiting
- ☐ Other (Specify) Contact R.P.
- ☐ Survey Meter Required in HRA's
- ☐ High Rad Door Watch Required

1. dpm/100cm²	5. dpm/100cm²	9. dpm/100cm²	13. dpm/100cm²	17. dpm/100cm²
2. dpm/100cm²	6. dpm/100cm²	10. dpm/100cm²	14. dpm/100cm²	18. dpm/100cm²
3. N/R dpm/100cm²	7. N/R dpm/100cm²	11. N/R dpm/100cm²	15. N/R dpm/100cm²	19. N/R dpm/100cm²
4. dpm/100cm²	8. dpm/100cm²	12. dpm/100cm²	16. dpm/100cm²	20. dpm/100cm²

Comments: 599 Rx Vessel Core Bore Holes after Neutron Window Removal

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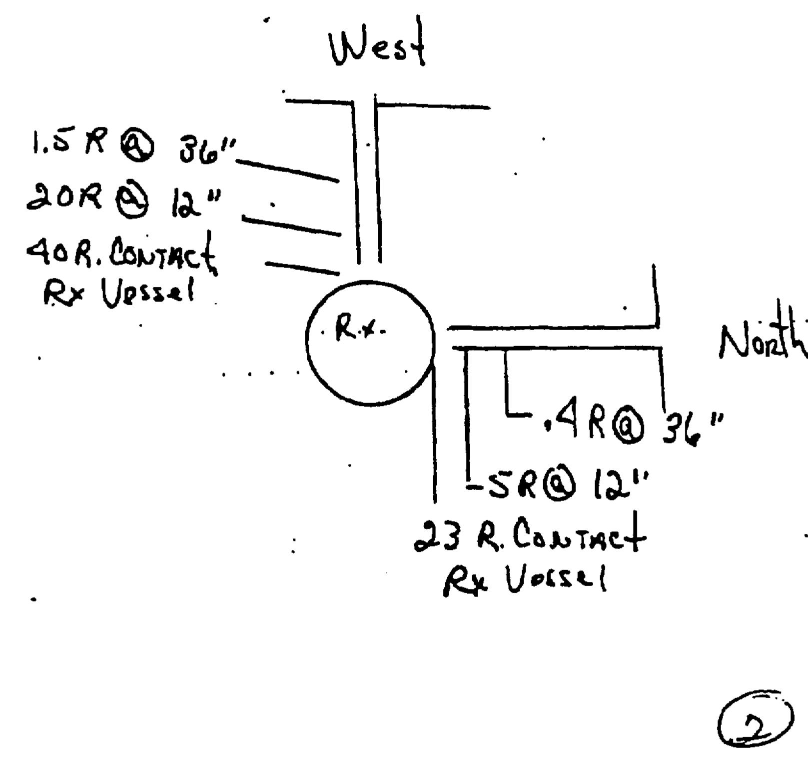
Map Key: #: General Area Dose Rate C/GA: Contact and General Area Dose Rate ⓢ: Smear Location --x--x--: Radiological Boundary LDWA: Low Dose Waiting Area *: Hot Spot ▲: Air Sample Location



Unposted Hot Spots May be Present

Ops Rounds
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

RP Surveys
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

Tours/Inspections
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

Work Entries
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R.

Dose Rate Meter/SN: Tele 29667	Cal Due: 7/17/00	Air Sampler: N/R	Cal Due: N/R
Counting Inst./SN: N/R	Cal Due: N/R	Inst. Type/SN: N/R	Cal Due: N/R.
Performed By: J. Kurtz	Date: 6-7-00	Reviewed By: [signature]	Date: 6-8-00

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RADIOLOGICAL SURVEY AND STATUS SHEET

UFI: 24*11*01

Survey Log # 2000-941

Date/Time: 7-28-00 1000 Technician: Kunz	Location: Sphere 5991
High Dose Rate: 40R/hr mRem/hr(γ) N/R mRem/hr(β)	General Dose Rate: N/R mRem/hr(γ) N/R mRem/hr(β)
High Contamination: N/R dpm/100cm²	General Contamination: N/R dpm/100cm²
Airborne Concentration: N/R µCi/cc	Spectrum No: N/R

Area Posting:
- ☒ Contamination Area (CA)
- ☐ High Radiation Area (HRA)
- ☐ High Contamination Area (HCA)
- ☐ Radiation Area (RA)
- ☐ Locked High Radiation Area (LHRA)
- ☐ Other (specify) ____
- ☐ Airborne Area (ABN)

Special Instructions:
- ☒ Notify Rad Pro PRIOR to any "Work Entry"
- ☐ Air Sampling Required for Entry
- ☐ Notify Monitoring Station When Entering and Exiting
- ☐ Other (Specify) ____
- ☐ Survey Meter Required in HRA's
- ☐ High Rad Door Watch Required

1. ____ dpm/100cm²	5. ____ dpm/100cm²	9. ____ dpm/100cm²	13. ____ dpm/100cm²	17. ____ dpm/100cm²
2. N/R dpm/100cm²	6. ____ dpm/100cm²	10. ____ dpm/100cm²	14. ____ dpm/100cm²	18. ____ dpm/100cm²
3. N/R dpm/100cm²	7. N/R dpm/100cm²	11. N/R dpm/100cm²	15. N/R dpm/100cm²	19. N/R dpm/100cm²
4. ____ dpm/100cm²	8. ____ dpm/100cm²	12. ____ dpm/100cm²	16. ____ dpm/100cm²	20. ____ dpm/100cm²

Comments: Dose Rates Take w/ Teletector through core bore holes on Sphere laydown and RM 441

Map Key: #: General Area Dose Rate C/GA: Contact and General Area Dose Rate ①: Smear Location --x--x--: Radiological Boundary LDWA: Low Dose Waiting Area *: Hot Spot ▲: Air Sample Location

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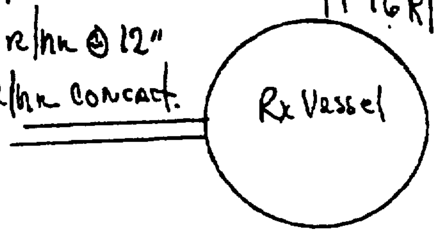
North Side.
400mrem/hr @ 36"
5R/hr @ 12"
16R/hr contact

West Side
5R/hr @ 36"
20R/hr @ 12"
45R/hr contact

W



3

Unposted Hot Spots May be Present

Ops Rounds
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

RP Surveys
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

Tours/Inspections
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

Work Entries
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/R

Dose Rate Meter/SN: Tele # 59805	Cal Due: 1/18/01	Air Sampler: N/R	Cal Due: N/R
Counting Inst./SN: N/R	Cal Due: N/R	Inst. Type/SN: N/R	Cal Due: N/R
Performed By: J. Kunz	Date: 7-28-00	Reviewed By: [signature]	Date: 7-28-00

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RADIOLOGICAL SURVEY AND STATUS SHEET

UFI: 24*11*01 1-17-01

Survey Log # 2001-1584

Date/Time: 1/17/01 Technician: E. Kurtz Location: Rm 441

High Dose Rate: 18000 mRem/hr(γ) N/A mRem/hr(β)
General Dose Rate: ≤1 mRem/hr(γ) N/A mRem/hr(β)

High Contamination: N/A dpm/100cm²
General Contamination: N/A dpm/100cm²

Airborne Concentration: N/A µCi/cc
Spectrum No: N/A

Area Posting:
- ☒ Contamination Area (CA)
- ☐ High Radiation Area (HRA)
- ☐ High Contamination Area (HCA)
- ☐ Radiation Area (RA)
- ☐ Locked High Radiation Area (LHRA)
- ☐ Other (Specify)
- ☐ Airborne Area (ABA)

Special Instructions:
- ☒ Notify Rad Pro PRIOR to any "Work Entry"
- ☐ Air Sampling Required for Entry
- ☐ Notify Monitoring Station When Entering and Exiting
- ☐ Other (Specify)
- ☐ Survey Meter Required in HRA's
- ☐ High Rad Door Watch Required

1. N/A dpm/100cm²	5. N/A dpm/100cm²	9. N/A dpm/100cm²	13. N/A dpm/100cm²	17. N/A dpm/100cm²
2. dpm/100cm²	6. dpm/100cm²	10. dpm/100cm²	14. dpm/100cm²	18. dpm/100cm²
3. dpm/100cm²	7. dpm/100cm²	11. dpm/100cm²	15. dpm/100cm²	19. dpm/100cm²
4. dpm/100cm²	8. dpm/100cm²	12. dpm/100cm²	16. dpm/100cm²	20. dpm/100cm²

Comments: DOSE RATE VERIFICATION @ N. CORE BORE RM 441
AREA SECURED W/ BOLTED BOARD

Map Key: #: General Area Dose Rate C/GA: Contact and General Area Dose Rate ①: Smear Location
--x--x--: Radiological Boundary LDWA: Low Dose Waiting Area *: Hot Spot Δ: Air Sample Location

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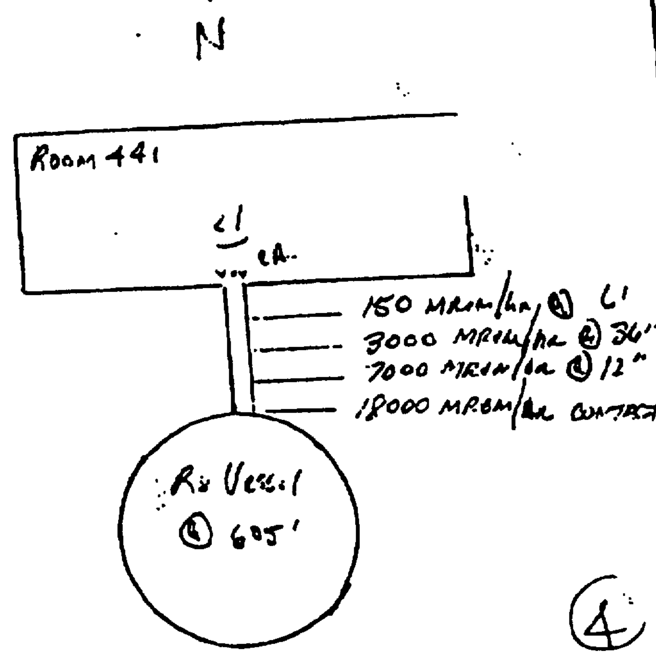

Unposted Hot Spots May be Present

Ops Rounds
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/A

RP Surveys
- ☒ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

Tours/Inspections
- ☐ Gloves for reach ins
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/A

Work Entries
- ☐ Class A
- ☐ Class B
- ☐ Class C

N/A

Dose Rate Meter/SN: Tele 59805	Cal Due: 7/15/01	Air Sampler: N/A		Cal Due: N/A
Counting Inst./SN: N/A	Cal Due: N/A	Inst. Type/SN: N/A		Cal Due: N/A
Performed By: E. Kurtz	Date: 1/17/01	Reviewed By: Sergio Iparin		Date: 1-17-01

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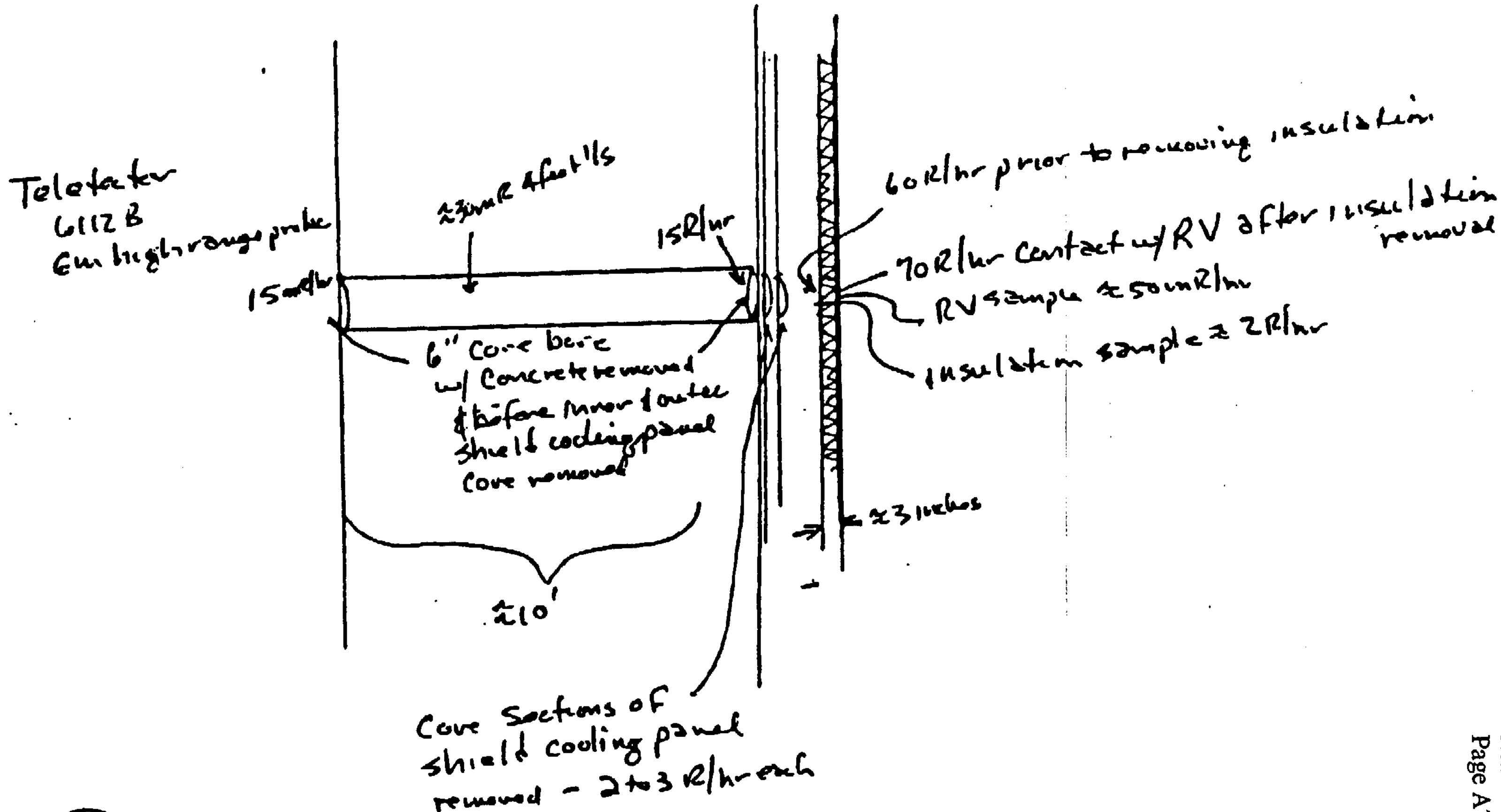
INFORMATION COPY



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ATTACHMENT B
ISOSHLD-PC/QAD Input/Output Files

A listing of the directory which contains the ISOSHLD-PC/QAD executable files follows. The PC used to run the code for this evaluation is also identified.

```
 Volume in drive C is PC5597
 Volume Serial Number is 3311-10DC
 Directory of C:\BigRock\10525-Rev1\RUNTEx\ex

.              <DIR>        11-01-00  7:29a .
..             <DIR>        11-01-00  7:29a ..
ISDRTP   LIB       232,018  08-13-91 11:39a ISDRTP.LIB
ISOSHLD  EXE     1,056,526  02-29-00  2:17p ISOSHLD.EXE
2 file(s)        1,288,544 bytes
         2 dir(s)      78,741,504 bytes free
```

A list and description of the ISOSHLD-PC and QAD output files is contained in the table that follows. Note that the output files contain a complete listing of the input data. As such only the output files are included in the table. The input files have the same name as the output files except they end in 'dat' instead of 'txt'.

All of the ISOSHLD-PC and QAD input and output files that were used in this evaluation are included in the CD which accompanies this calculation.

ISOSHLD-PC/QAD Files		
Number	File Name	Discription
QAD Runs		
1	RVCONAC.txt	Determines contact and 1 meter dose rates from reactor vessel insulation outer surface, along the active core midplane, due to reactor vessel source terms in the active core region.
2	INCONAC.txt	Determines contact and 1 meter dose rates from reactor vessel insulation outer surface, along the active core midplane, due to reactor vessel insulation source terms in the active core region.
3	TSCONAC.txt	Determines contact and 1 meter dose rates from reactor vessel insulation outer surface, along the active core midplane, due to thermal shield source terms in the active core region.

ISOSHLD-PC/QAD Files		
Number	File Name	Discription
4	SWCONAC.txt	Determines contact and 1 meter dose rates from reactor vessel insulation outer surface, along the active core midplane, due to thermal shield seal weight source terms in the active core region.

FINAL

CALCULATION N-10902-010-001

REVISION 1

ATTACHMENT “C” NOT INCLUDED

APPENDIX 5-4

WASTE CHARACTERIZATION

I. Summary

Radionuclide activity totaling approximately 425 Curies, representing about 93% of the estimated original activity present on piping and about two thirds of the activity originally present in the reactor vessel, steam drum and heat exchangers, was removed from the reactor water, shutdown cooling and cleanup systems by the chemical decontamination process. Waste quantities, (activities and volumes) waste characterization (including transuranic levels) and other chemical decontamination waste considerations are discussed in this chapter.

II. Radionuclides Characterization

Radioactive waste produced by the DFD decontamination process was 660 cubic feet of ion exchange resin. This total included 540 ft^3 of cation resin, 90 ft^3 of anion resin and 30 ft^3 of mixed bed resin. Of this total approximately 110 ft^3 or 17 percent was generated by removal of process chemical. The remaining 83 percent was generated by removal of metals dissolved during the decontamination. Additional waste consisted of 150 spent filter (30 inch) cartridges from the process skid, plus some contribution to plant resins and filters caused by a plant system check valve leak. This leak also contributed to higher than projected resin generation due to high temperature decontamination fluid contact with carbon steel downstream of the check valve during the stainless steel cycle.

Although only 300 Ci of gamma emitters were expected to be removed, actually approximately 400 Ci on resins, and approximately 25 Ci on filters were detected. This value are close to the shipping limits of about 0.8 Ci/ft^3 of gamma emitters (mostly Co-60) in resins, and 400 Ci in 660 ft^3 put us at a respectable 0.6 Ci/ft^3. This meant that the normal shipping methods (typically using CNS-14-190 casks) remained suitable for the decon resins. Filters also were easily handled, primarily because change-out was based on 10 rem/hr maximum contact rate.

Fuel failures early in the operating history of Big Rock Point led to concerns that transuranic activity could have remained within the system under layers of more recently deposited activation products, and that these radionuclides could be released in significant quantities as a result of aggressive chemical decontamination. Both sampling and *In Situ* gamma spectroscopy were utilized during the chemical decontamination process to track the radionuclide components removed from the system and deposition in low flow areas.

Emphasis was placed on detection of Am-241 by means of its 59.5 KeV photopeak as an indication of transuranic presence. Earlier studies had shown Am-241 to be an excellent indicator of transuranic presence. Scaling of transuranics normally is made to Co-60. Graph VII-1 shows the relationship of Co-60 to Pu-239. In contrast, Graph VII-2 shows the much better correlation of Am-241 to Pu-239. Both figures represent the last nine years of Co-60, Am-241 and Pu-239 data from 10 CFR Part 61 scaling factor analyses for Big Rock Point radioactive waste streams.

Early detection of the 59.5 KeV photopeak by *In Situ* analysis during the first cycle of chemical decontamination caused concern that the transuranics had indeed been enhanced by the decontamination process. However, it was quickly determined that Ta-182, from hafnium control blade activation, was responsible for the 59.5 KeV peak (Ta-182 has companion peaks at 68, 100, 1,121 and 1,221 KeV, and gamma yields accounted for all the activity at 59.5 KeV as due to Ta-182). In addition, alpha counts of coolant and filter solids demonstrated the lack of any significant transuranic activity. The half-life of Ta-182 at 115 days is relatively short compared to Am-241. Tests on artifacts with long decay had not shown the presence of Ta-182 and thus had not interfered with Am-241 analyses.

Graph VII-1 & VII-2, represent pre and post-decontamination scaling ratios, and demonstrate how well the decontamination media points (triangles) fit within the scatter of pre-decontamination data.

Figure VII-1 Relationship of Co-60 to Pu-239

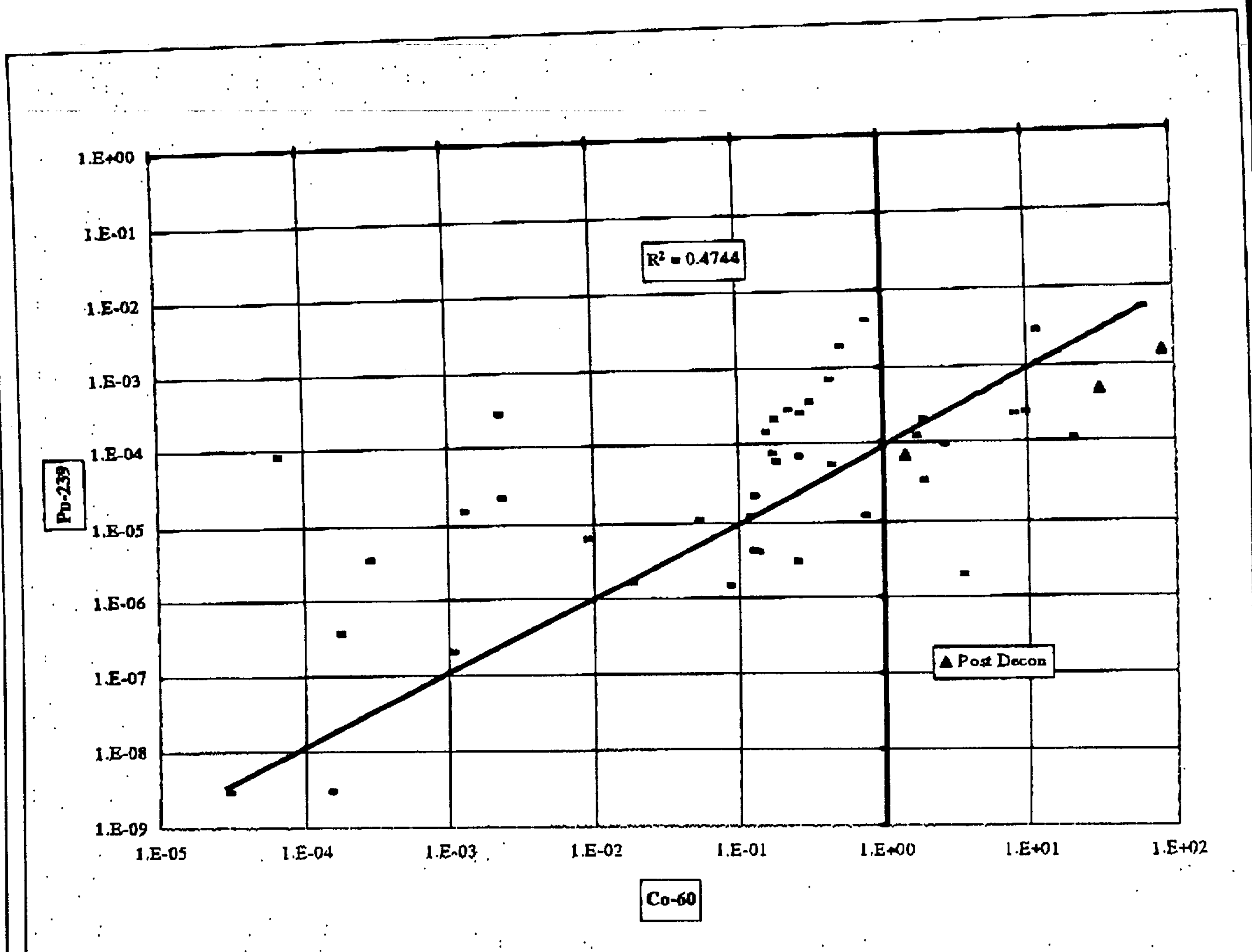
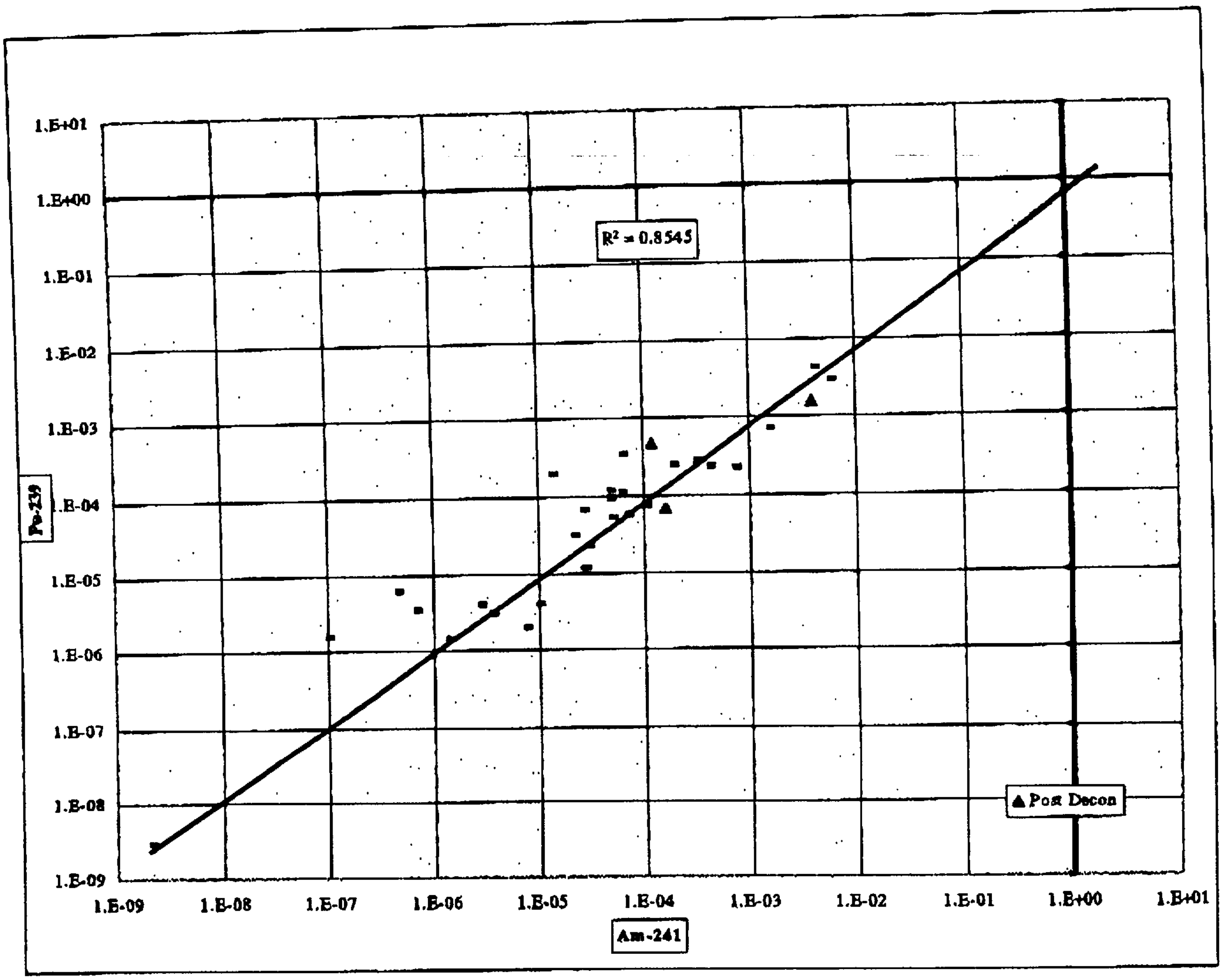


Figure VII-2 Correlation of Am-241 to Pu-239



III. Radionuclide Scaling Factor

Samples of coolant, resins, filter sludge and dry active waste accumulated during the chemical decontamination process were evaluated for hard-to-detect radionuclides (10 CFR Part 61 scaling factor analysis) by a contract laboratory. Tables VII-1 through VII-4 indicate that median and mean scaling factor ratios of decontamination samples to historic scaling factors are near unity (plus or minus a factor of about 4). Since historic scaling factors typically vary by the same or larger amount, this indicates lack of any significant change in ratios due to chemical decontamination.

Tables VII-1 through VII-4 Median & Mean scaling factor ratios of decontamination samples to historic scaling factors.

Table VII-1 Chem Decon Dry Waste Nuclide Ratios

Nuclide	Observed (µCi/g)	Observed /Co-60*	Historical /Co-60	Ratio New/Old
Fe-55	7.8E-02	5.5E-02	3.3E-01	1.7E-01
Ni-63	1.2E-02	8.5E-03	8.0E-02	1.1E-01
Co-60	1.4E+00	1.0E+00	1.0E+00	1.0E+00
Sr-90*	1.1E-03	2.0E-02	6.4E-02	3.1E-01
Am-241	1.6E-04	1.1E-04	1.3E-04	8.9E-01
Cm-242	4.9E-06	3.5E-06	9.1E-06	3.8E-01
Pu-238	1.5E-05	1.1E-05	5.0E-05	2.1E-01
Pu-239/40	7.0E-05	5.0E-05	1.4E-04	3.6E-01
Cs-137*	5.6E-02			
			median	0.333
*Ratio for Sr-90 is /Cs-137			mean	0.429

Table VII-2 Chem Decon Filter Nuclide Ratios

Nuclide	Observed (µCi/g)	Observed /Co-60	Historical /Co-60	Ratio New/Old
Fe-55	7.5E+00	9.0E-01	3.3E-01	2.8E+00
Ni-63	5.9E-01	7.1E-02	4.4E-01	1.6E-01
Co-60	8.3E+00	1.0E+00	1.0E+00	1.0E+00
Am-241	3.9E-03	4.7E-04	1.3E-04	3.7E+00
Cm-242	2.6E-04	3.1E-05	9.1E-06	3.4E+00
Cm-243/44	2.0E-04	2.4E-05	7.2E-06	3.3E+00
Pu-238	6.7E-04	8.1E-05	5.0E-05	1.6E+00
Pu-239/40	1.5E-03	1.8E-04	1.4E-04	1.3E+00
			median	2.192
			mean	2.167

Table VII-3 Chem Decon Coolant Nuclide Ratios

Nuclide	Observed (µCi/g)	Observed /Co-60*	Historical /Co-60	Ratio New/Old
Ni-59	1.1E-06	1.6E-03		
Tc-99	1.6E-06	2.3E-03	4.8E-03	4.7E-01
Fe-55	3.4E-04	4.8E-01	3.3E-01	1.5E+00
Ni-63	1.3E-05	1.8E-02	4.4E-01	4.2E-02
Sr-90*	3.5E-07	2.9E-02	6.4E-02	4.6E-01
Co-60	7.0E-04	1.0E+00	1.0E+00	1.0E+00
Cs-137*	1.2E-05			
			median	0.474
*Ratio for Sr-90 is /Cs-137			mean	0.691

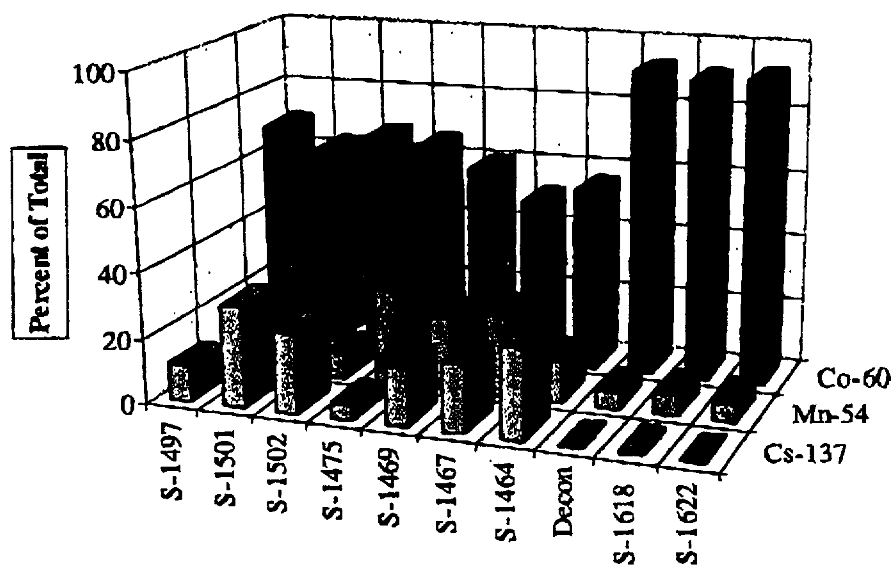
Table VII-4 Chem Decon Resin Waste Nuclide Ratios

Nuclide	Observed (µCi/g)	Observed /Co-60	Historical /Co-60	Ratio New/Old
Fe-55	3.6E+01	1.2E+00	3.3E-01	3.6E+00
Ni-63	3.5E+00	1.1E-01	4.4E-01	2.6E-01
Co-60	3.1E+01	1.0E+00	1.0E+00	1.0E+00
Am-241	1.2E-04	3.9E-06	1.3E-04	3.0E-02
Cm-242	1.6E-04	5.2E-06	9.1E-06	5.7E-01
Pu-238	3.7E-04	1.2E-05	5.0E-05	2.4E-01
Pu-239/40	4.7E-04	1.5E-05	1.4E-04	1.1E-01
			median	0.258
			mean	0.822

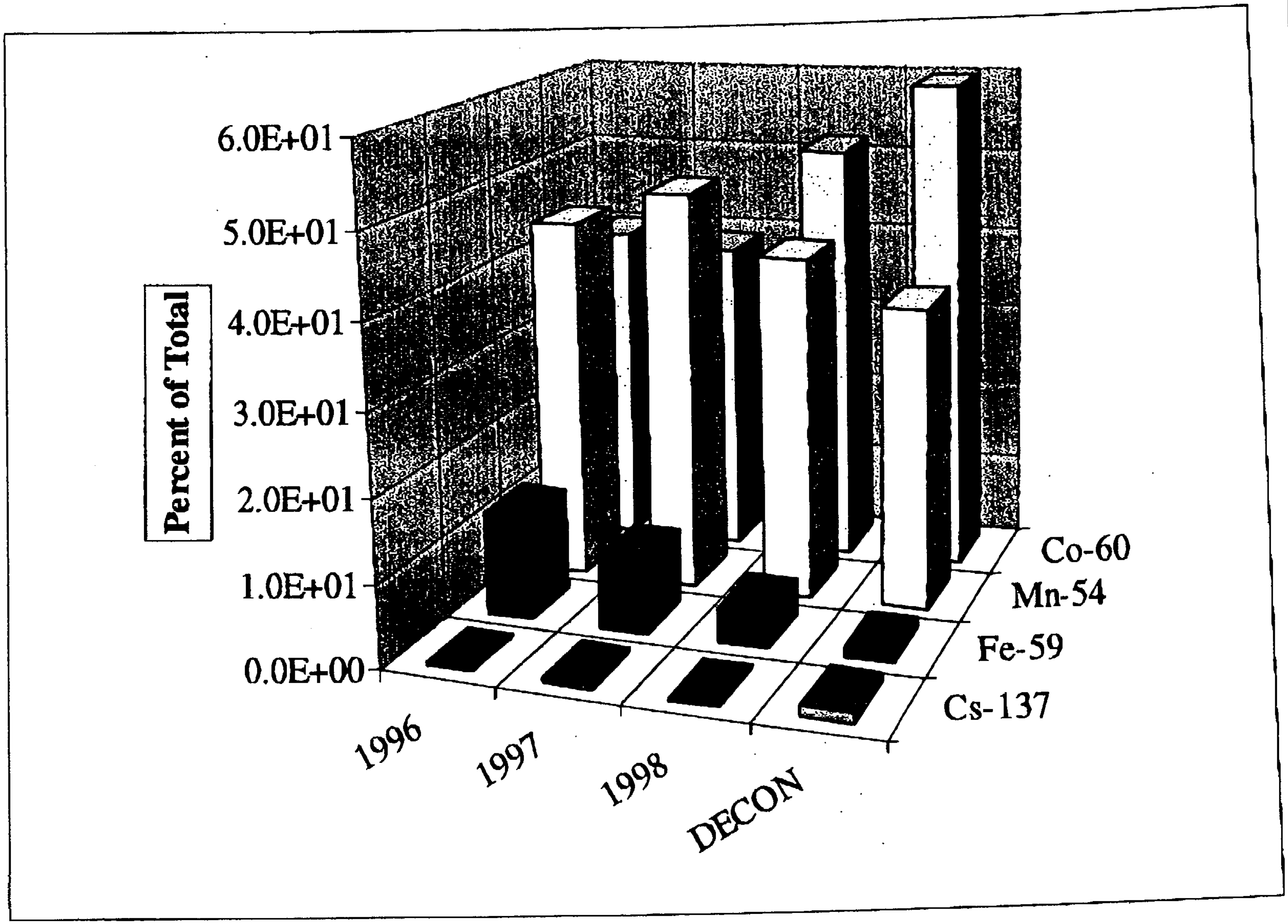
IV. Nuclides in Resin and filters

The gamma nuclides shown in Graph VII-3, & VII-4 (resins and filters, respectively) show that levels of Co-60 increased due to chemical decontamination, while the shorter half-life nuclides, Fe-59 and Mn-54, decreased. This effect is due to removal of older deposits during the decon process which were enhanced in Co-60 relative to Mn-54 based on the longer half-life of Co-60, and to decay since shutdown in August, 1997. Lower power levels during the long coast-down in power during the last seven months of operation also contributed. Note that Figure 4 shows that in 1998 prior to chemical decontamination, Mn-54 and Fe-59 in filters had already been lowered. Percentage values for Cs-137 also decreased in resins, but that is because the resin storage tanks held a mixture of condensate system resins and radwaste resins, with condensate system resins accounting for most of the Cs-137 in the older samples.

Figures VII-3 Resins levels of Co-60 increased due to chemical decontamination



Figures VII-4 Filters levels of Co-60 increased due to chemical decontamination



CONCLUSION & LESSONS LEARNED

I. Summary

Chemical decontamination with the *DFD* process was highly successful at Big Rock Point. Dose rates to personnel involved in decommissioning activities have been reduced by more than a factor of 12 from pre-decontamination levels, and average contact readings with pipes and equipment have been reduced by greater than a factor of 25. Reduction in dose rates due to system decontamination has allowed additional detection of sources not previously identifiable due to high system dose rate contributions. Detection and elimination of hot spots and other contributors such as activity associated with certain floor areas, drains and sumps, have reduced personnel doses another factor of two in some of the previously highest dose rate areas.

Radioactivity removed from plant systems was well within normal handling capabilities. Demineralizer resins at Big Rock Point accumulated during power operation for several years prior to shipment for burial. At the time of chemical decontamination, resins had been accumulated for about two years. With the addition of 660 ft^3 of waste generated by chemical decontamination, the storage was still within the normal storage capacity. Filter activity was limited by change-out at or below a preset contact dose rate of 10 rem/hr which was considerably lower than many of the operating plant filters. Controlled in this manner, decontamination filters presented no handling problems.

Decontamination did not result in any observed transuranic increase within the radioactive waste streams. The changes which did occur, such as a slight increase of Co-60 and a decreased of Mn-54, Fe-59 and Cs-137 contributions to total gamma radioactivity in resins, have not caused any significant radwaste handling or shipping problems.

II. Project Overview Analysis

The entire decontamination process effort at BRP was successful in that quantifiable DFs were obtained and all decon operations were performed within the allotted time schedules. No major problems were encountered during the decontamination. The following is the brief analysis of the project.

1) System set up went smoothly, except for a short delay due to a minor leak but it did not appear to affect the rate of oxide removal or decontamination factors obtained on the primary system.

2) Preparing all necessary plant documentation well ahead of time helped to ensure smooth applications.

3) Contingencies were planned for, such as adequate chemicals for an additional decontamination step or adequate chemical residency time greater than what was actually planned in order to offset any unforeseen problems during the actual application. A preparation meetings was held ever shift and expected process and execution times were reviewed and proper work departments notified of the schedule

4) Pre-operational inspections were rigorously applied and all steps of the set-up were included in a check-list.

5) Briefings were held for each major work evolution. This approach provided specific information relative to the current radiological conditions, and assisted in limiting decontamination personnel dose to 17.25 person-rem, well below the projection of 25 person-rem.

6) The most effective exposure reduction technique observed during the Chem Decon Process was simply restricting access to those individuals necessary to the safe operation of the PN Services equipment. All loitering and job site inspections were kept to a minimum as the Decontamination Managers provided routine de-briefings and current status report.

7) In addition to the above, cuno filter replacement performed without remote tool also proved to be quite effective. This concept was initially met with curious resistance but later proved to be an efficient means of transfer. As recorded by video, the first filter change out was performed in < 45 seconds thus expending only 0.04 person-rem. Experienced personnel performed this evolution. A mock-up would have been used to improve efficiency, for inexperienced workers.

8) Health Physics provided adequate radiological controls in preparation for unknown conditions. Much emphasis was placed on restricting non-essential personnel from the work area, making routine surveillance to monitor changing dose rates, and providing 24 hour coverage

9) The airborne radiological controls as outlined in the RWP and specific to the pre-job brief provided positive radiological controls at all times.

III. Lessons Learned

1) To determine the DF factors at pre-selected locations a Juno type ion chamber was used for surveying the locations. The ion chamber was calibrated to CS-137 due to its long half-life. This ion chamber is a proper instrument to set dose rate requirements for personnel performing an activity in a controlled area. However, due to its chamber configuration it is very difficult to pin point a particular hot spot, especially a crud trap within a high background. To overcome this problem a bi-directional shielded probe such as an Eberline E530-N should be used. This instrument is very useful to detect any hot spots within high background radiation area(s). The instrument should be calibrated to Co-60, since this isotope is the predominant radionuclide before and after the decontamination process.

2) An alternative to enhance the decontamination factor is to hydrolyze the system after the decontamination process. However, the system does not have any special flange or location to be able to perform the hydrolyzing. The average DF could be significantly enhanced by hydrolyzing the system, thereby removing the particulate material having a high activity concentration that is settled out in low areas of the system.

3) There were several incidences over the duration of the Project that could have been avoided if more emphasis had been put on planning. The following are points of interest and proposed corrective actions:

   a) The cuno filters was not loaded on schedule. As a result of this oversight, final preparations and equipment set up was performed in a much higher dose field.

   b) Prior to the sample collection, working area dose rates were approximately 2 mr/hr and increased to 60 mr/hr, due to accumulation of activity on ion exchange column. This accounts for nearly 2000 mr of unnecessary and unplanned exposure.

   c) Cuno filter change out and resin generation exceeded initial scope

4) The 599' elev. of the Sphere was used as the main PN Services lay down area. Because of the space restrictions, it was difficult to properly shield certain components. (i.e. F-69 Filter housing, heat exchanger, mix tank, etc.) If this process could have been set up using more floor space, better and more effective shielding could have been installed lowering personnel exposure.

CONCLUSION & LESSONS LEARNED

Decontamination for Decommissioning

5) Several engineering controls were implemented throughout this process. The use of remote Area Radiation Monitor's (ARM) gave continuous readings at specified locations. Teledosimetry and cameras also were of benefit in man-rem savings. HEPA ventilation provided negative pressure and regulated air flow during chemical addition evolution, however the ventilation was not beneficial during cuno filter exchange, while they were wet.

6) Although remote readings were obtainable with Teledosimetry, no real confidence was put into their operability or accuracy. The cable lengths ordered did not match those received and thus resulted in longer set up times and more frequent High Radiation Area entries to verify dose rates and DF (decon factor).

7) Although many individuals were dedicated full time to the project, all personnel should have been dedicated.

8) The following items should be incorporated into the pre-planing process:

   a) Planning for all potential contingency scenarios.
   b) Installing a self-check mechanism to ensure lock-out/tag-out system and valve line-up's are correct.
   c) Allowing enough space to properly shield components.
   d) Using standing water shields to provide more effective low dose waiting areas (LDWA's) than traditional lead racks or booths.
   e) Controlling proper FME zone and parts accountability to save exposure by limiting High Rad entries.
   f) Evaluating usage of electric or pneumatic tools for Filter Housing bolts, rather than manual detensioning, which takes longer.
   g) Replacing cuno filters properly trained personnel (dry runs and mockups) should be considered.

9) Hydrolansing should be planned for (as a contingency), based on the radiation survey data after the decontamination. Note: Hydrolansing will help to remove any dead end deposits and precipitation that are present after the decontamination, and thereby will result in a higher decontamination factor.

IV. REFERENCES

1) NUREG/CR-4445 "Effectiveness and Safety Aspects of Selected Decontamination Processes" August 1986

2) Electric Power Research Institute (EPRI) report TR-106386-3500-28 *Decontamination for Decommissioning* May 1996

3) Electric Power Research Institute, *The EPRI DFD Process: Decontamination for Decommissioning*. Nuclear Power Group Technical Brief, June, 1997

4) Robert A. English, Richard L. Burdette and William E. Kessler, "The New Radiological Criteria for Decommissioning: Importance of Transuranic Nuclides", DDER-'94, Vol 1, pp 253-258, ANS Topical Symposium, Nov. 13-18, 1994.