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SOUTHERN COMPANY
Energy to Serve Your World℠

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Fluence Calculations and Methodology for the
Unit 1 Updated Analysis of Core Shroud Vertical Welds

Ladies and Gentlemen:

By letter dated November 14, 2003, Southern Nuclear Operating Company (SNC) provided to the NRC the results of the updated analysis of the Edwin I. Hatch Nuclear Plant Unit 1 flawed Core Shroud Vertical welds.

Subsequently, during a phone conversation in December 2003, the NRC requested a copy of the calculation and a description of the methodology. A copy of the calculation is enclosed. This Hatch Unit 1- specific calculation is not considered proprietary. However, the methodology, which is referenced in the enclosed calculation, is described in BWRVIP proprietary documents previously transmitted to the NRC. Those documents are listed below.

As noted in the NRC Agencywide Documents Access and Management System (ADAMS), the *BWRVIP-115: BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems* (reference 3 of the enclosed calculation) was transmitted to the NRC on June 26, 2003 (Accession Number ML 031820253). The *BWRVIP-117: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5* (reference 4 of the enclosed calculation) was transmitted to the NRC on August 5, 2003 (Accession Number ML 032230313). The *BWRVIP-114: BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual* was transmitted on June 11, 2003 (Accession Number ML 031640194). An Errata to *BWRVIP-114* was sent on August 21, 2003 (Accession Number ML 032380140). Additionally, the *BWRVIP-121: BWR Vessel and Internals Project, RAMA Fluence Methodology Procedures Manual* was transmitted on October 29, 2003 (Accession Number ML 033040182).

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This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

H. L. Sumner, Jr.

HLS/whc/daj

Enclosure: TransWare Report Number TWE-HATCH1-001-R-001, Revision 0

cc: Southern Nuclear Operating Company
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Document Services RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
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EVALUATION OF >1.0 MEV FLUENCE IN THE HATCH 1 SHROUD

TransWare Report Number: TWE-HATCH1-001-R-001
Revision Number: 0

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Purpose

The purpose of the calculation documented in this report is to estimate the >1.0 MeV neutron fluence in the Hatch 1 shroud structure. The regions of interest in the shroud are locations where cracks have been documented in the Core Shroud Weld Identification Layout diagram [1].

Objective

The objective of the calculation is to estimate the >1.0 MeV neutron fluence at observed locations in the Hatch 1 shroud where cracking has been observed. The calculated fluence will be compared to the fracture toughness threshold of 1.0 x 10^{21} n/cm^2.

Model Description

For this study, a model shall be developed to calculate estimated neutron fluence values for energy >1.0 MeV at specified locations in the Hatch 1 core shroud structure. The locations of interest in the shroud are the azimuthal, elevation, and radial locations where cracks have been visually verified and documented in the Core Shroud Weld Identification Layout diagram [1]. Figures 1 and 2 illustrate the vertical and horizontal crack locations evaluated in this report. Figure 1 identifies the cracks in azimuthal angles 0 to 180 degrees and Figure 2 identifies the cracks in angles 180 to 360 degrees.

The neutron fluence calculations were performed using the RAMA Fluence Methodology software [2] that is under development by EPRI and the BWRVIP. The RAMA methodology has been developed and benchmarked [3, 4, 5] in accordance with the intent of U.S. NRC Regulatory Guide 1.190 [6]. The methodology has been submitted to the U.S. NRC for review. As of the date of this report, the methodology has not been approved for safety-related calculations.

The RAMA Fluence Methodology computer model assumes azimuthal symmetry in the reactor geometry. The reactor is represented by a north-northeast octant symmetric model that ranges from 0 to 45 degrees, where 0 degrees corresponds to the north compass direction specified in the reactor plan drawings. Figure 3 illustrates a top view of the reactor model cross section that corresponds to the core mid-plane axial elevation.

The computer model used in this analysis omits the regions outside the reactor pressure vessel, i.e., the cavity, mirror insulation, and biological shield regions. The assumption has no affect on the shroud calculation, as the downcomer region provides an adequate boundary condition for the analysis. The purpose of the assumption was used to allow the three-dimensional calculations to be performed within the allotted time for the analysis. When time permits, the model can be expanded to include these regions.

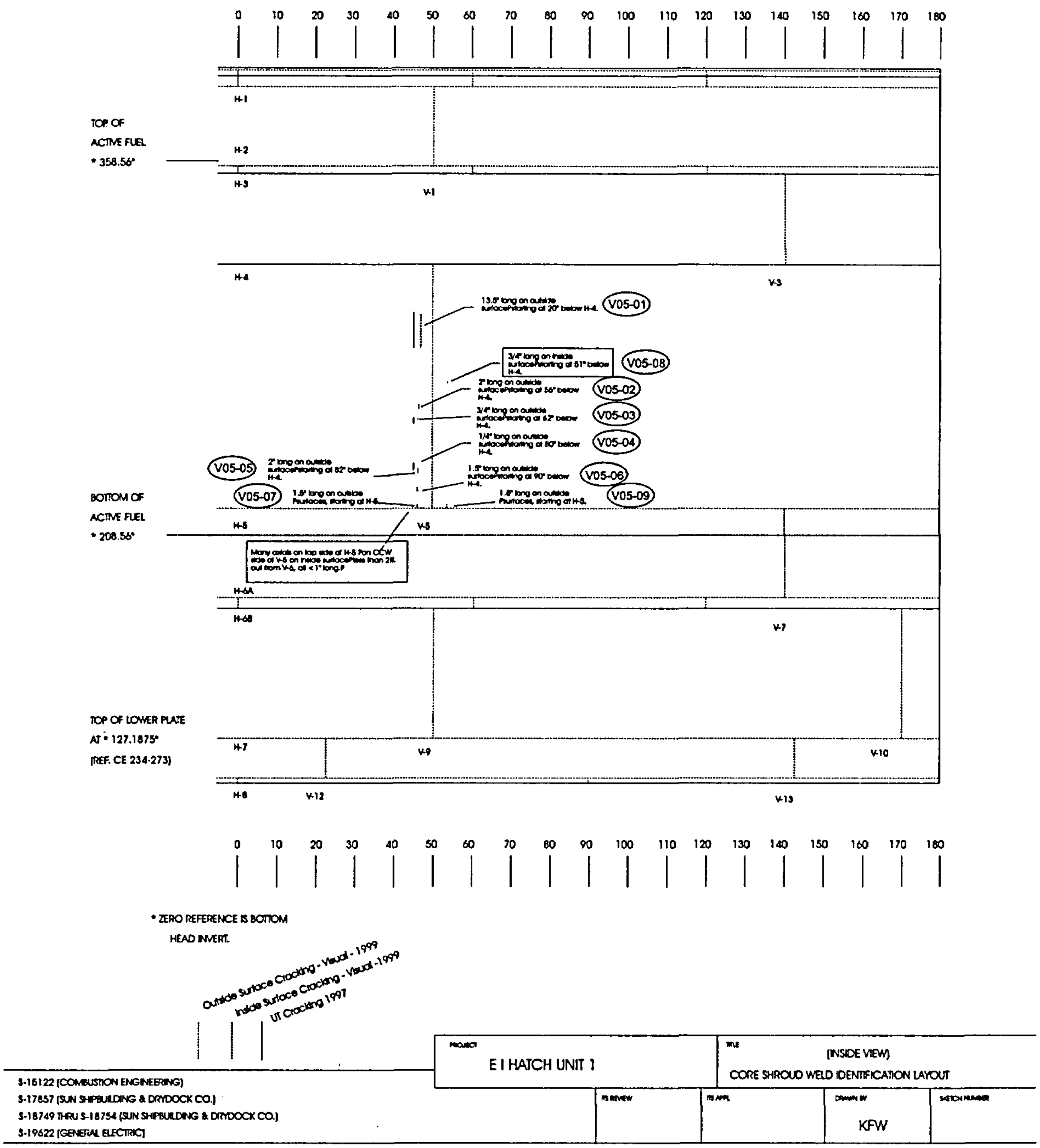


Figure 1
Illustration of Hatch 1 Shroud Cracks from Azimuths 0 to 180 Degrees

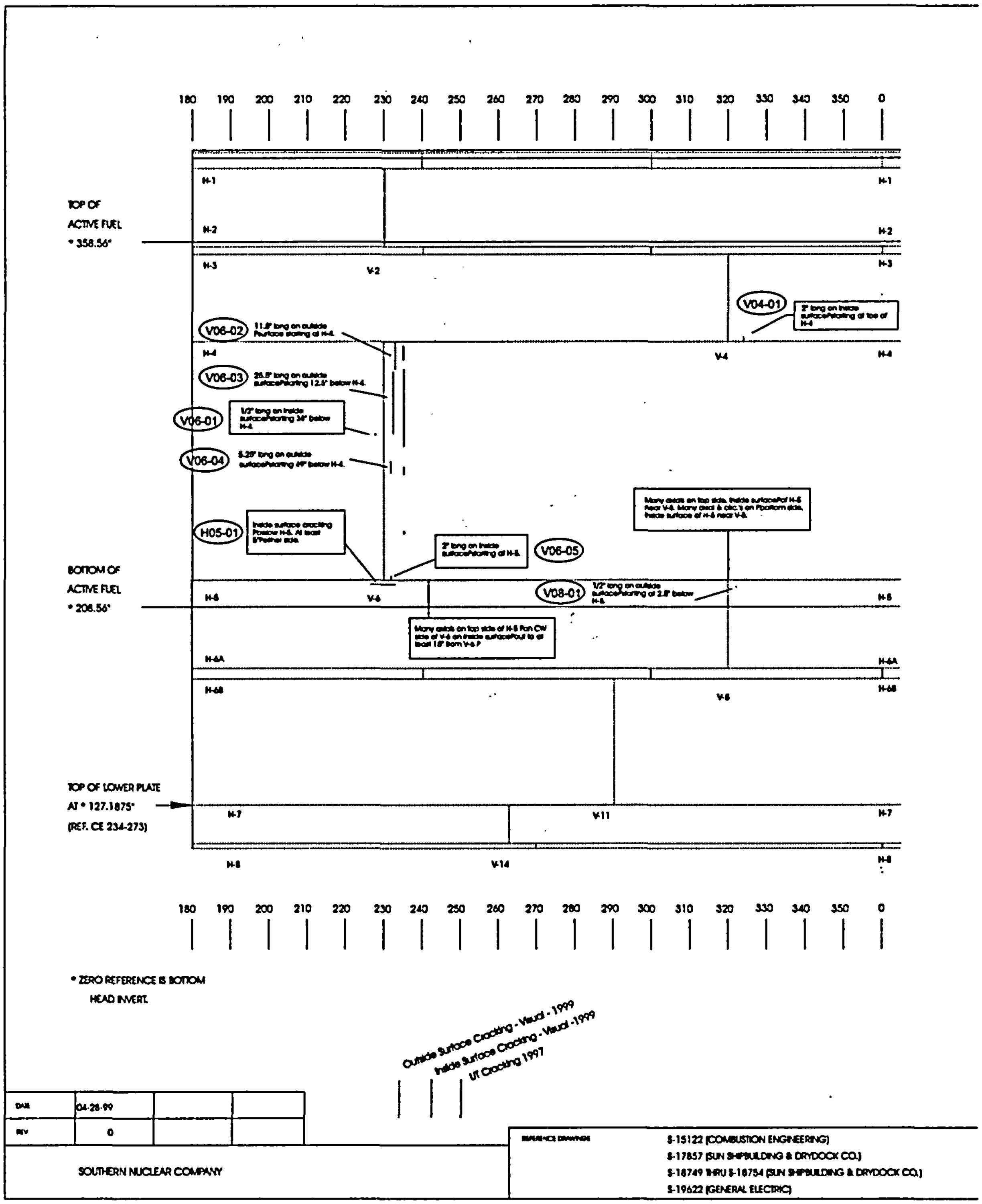


Figure 2
Illustration of Hatch 1 Shroud Cracks from Azimuths 180 to 360 Degrees

To do the analysis, the azimuthal positions of the cracks shown in Figures 1 and 2 were translated to the symmetrical azimuthal positions of the model that is illustrated in Figure 3. Table 1 identifies the cracks and provides the reactor actual and translated azimuthal angles for the model.

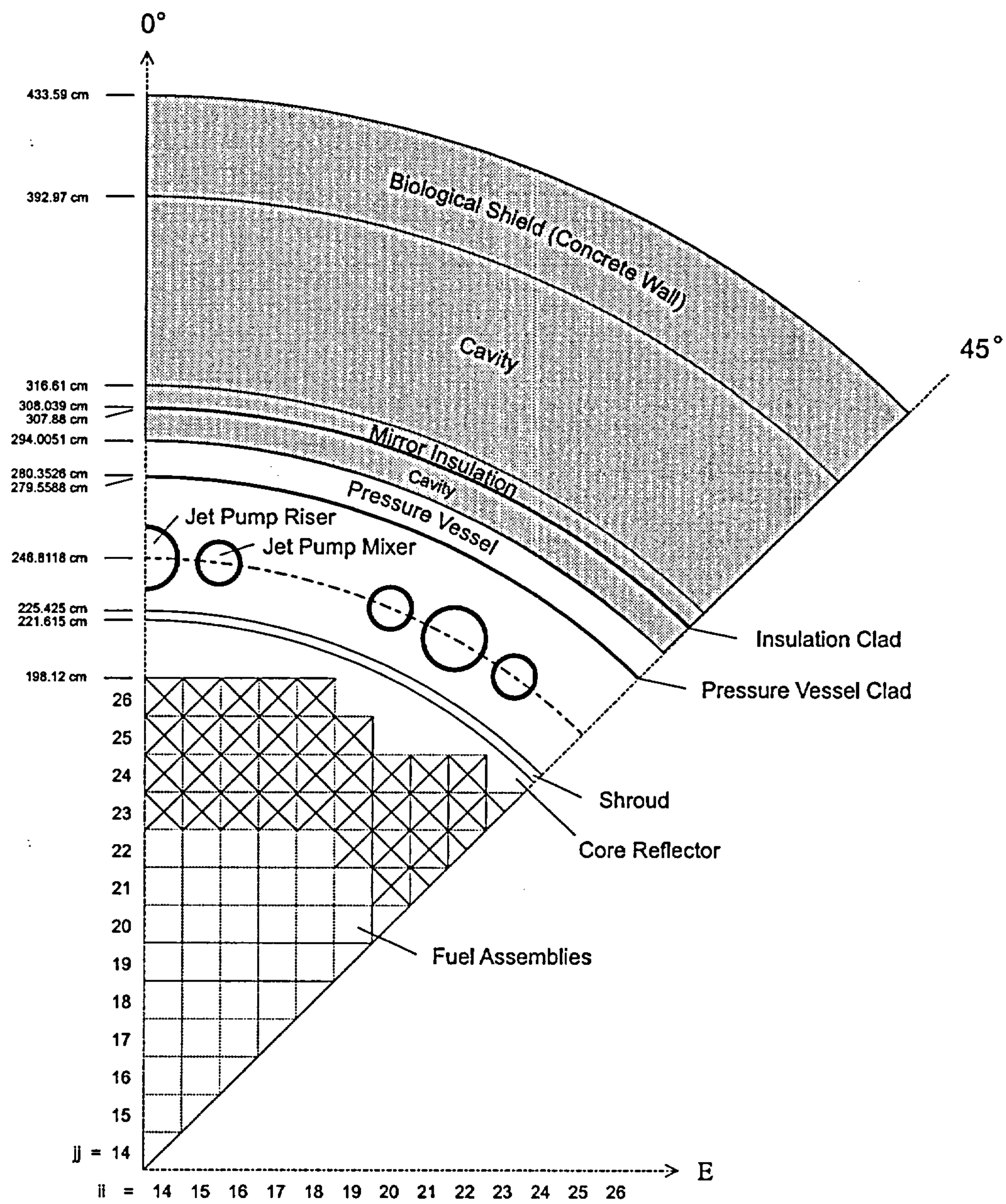


Figure 3
Top View of the Hatch 1 Computer Model

Table 1
List of Shroud Cracks

Crack ID	I/O	Azimuthal Locations Actual	Model	Length	Elevation (in cm)
V04-01 Inside 2"	I	324	36	5.08 cm (2.0")	807.415 - 812.495 cm (317.9 - 319.9")
V05-01 Outside 13.5"	O	47	43	34.29 cm (13.5")	722.325 - 756.615 cm (284.4 - 297.9")
V05-02 Outside 2"	O	47	43	5.08 cm (2.0")	660.095 - 665.175 cm (259.9 - 261.9")
V05-03 Outside ¾"	O	47	43	1.905 cm (0.75")	648.030 - 649.935 cm (255.1 - 255.9")
V05-04 Outside ¼"	O	47	43	0.635 cm (0.25")	603.580 - 604.215 cm (237.6 - 237.9")
V05-05 Outside 2"	O	47	43	5.08 cm (2.0")	594.055 - 599.135 cm (233.9 - 235.9")
V05-06 Outside 1.5"	O	47	43	3.81 cm (1.5")	575.005 - 578.815 cm (226.4 - 227.9")
V05-07 Outside 1.5"	O	47	43	3.81 cm (1.5")	554.025 - 557.835 cm (218.1 - 219.6")
V05-08 Inside ¾"	I	53	37	1.905 cm (0.75")	675.970 - 677.875 cm (266.1 - 266.9")
V05-09 Outside 1.5"	O	53	37	3.81 cm (1.5")	554.025 - 557.835 cm (218.1 - 219.6")
V06-01 Inside ½"	I	228	42	1.27 cm (0.5")	709.625 - 710.895 cm (279.4 - 279.9")
V06-02 Outside 11.5"	O	233	37	29.21 cm (11.5")	778.205 - 807.415 cm (306.4 - 317.9")
V06-03 Outside 25.5"	O	232	38	64.77 cm (25.5")	710.895 - 775.665 cm (279.9 - 305.4")
V06-04 Outside 5.25"	O	232	38	13.335 cm (5.25")	669.620 - 682.955 cm (263.6 - 268.9")
V06-05 Inside 2"	I	232	38	5.08 cm (2.0")	557.835 - 562.915 cm (219.6 - 221.6")
V08-01 Outside ½"	O	322	38	1.27 cm (0.5")	550.215 - 551.485 cm (216.6 - 217.1")
H05-01, Inside 10"	I	228.5 – 231.5	38.5 – 41.5	25.4 cm (10.0")	557.835 cm (219.6")

The reactor geometry and reactor operating data for the fluence calculation were taken from design inputs generated by Brookhaven National Laboratory (BNL) [5]. BNL generated the design inputs for the purpose of benchmarking computer methods against neutron fluence measurements that were taken from the Hatch 1 jet pump riser brace pads at the end of operating cycle 19. The operating data was specifically prepared to evaluate the brace pad measurements that were taken about 30 cm (12 inches) above the core mid-plane elevation. As a result, some of the operating data was prepared specifically to represent the operating conditions near the core mid-plane elevation. Following are discussions regarding modeling assumptions used in the jet pump riser brace pad model and how they might affect this analysis.

- The >1.0 MeV neutron fluence calculated for the Hatch 1 shroud is based on the RAMA Fluence Methodology computer model developed in [5]. The objective of the work was to calculate the >1.0 MeV neutron fluence at the jet pump riser brace pad locations. The brace pads are located at elevation 742.95 cm (292.5 inches), which is the centerline elevation for the pads. The pads are about 11.43 cm (4.5 inches) in height. The elevation of the brace pads is within about 30 cm (about 12 inches) of the core axial mid-plane, specified as elevation 712.6224 cm (280.56 inches). The results of the RAMA Hatch 1 benchmark show that the RAMA model over-predicted the >1.0 MeV fluence by about 17%. Thus, some conservatism exists in the current RAMA Hatch 1 model at the elevations near the jet pump riser brace pads.

- The jet pump riser brace pads are situated at about 30 cm (12 inches) above the core mid-plane elevation. This allowed certain assumptions to be made in the preparation of the Hatch 1 geometry model and reactor operating data. These assumptions are documented in [5]. Specifically, the reactor operating data for the Hatch 1 benchmark was prepared to reflect the operating conditions near the core mid-plane elevation. Operating data for off mid-plane elevations is not accurately represented. Consequently, the operating data that was prepared for the brace pad elevation is not fully applicable for the Hatch 1 shroud crack analysis that spans the full axial height of the reactor core region.

- The reactor operating history is modeled using 19 cycles of reactor operating data. Each cycle is characterized with one data set. This implies that a cycle-averaged, core axial power shape is used in the flux/fluence calculations. The axial power shape of a reactor can vary significantly throughout a cycle, therefore, the data used in this analysis may not provide an accurate representation of the core power for the shroud fluence evaluation. The affect of this approximation is not known.

- The core power distribution is grossly approximated for the first 12 operating cycles. Fairly detailed core power distribution data is provided for cycles 13 through 19. Following are discussions on the power distribution data used in the model:

  - The axial powers for the fuel assemblies in cycles 1 through 4 assumed that the active fuel height was 365.76 cm (144.0"). The axial powers for these cycles were provided in four 91.44 cm (36.0") axial segments.

- The axial powers for the fuel assemblies in cycles 5 through 12 assumed that the active fuel height was 381.0 cm (150.0"). The axial powers for these cycles were provided in three 91.44 cm (36.0") and one 106.68 cm (42.0") axial segments.

- The axial power data provided for cycles 1 through 12, which are represented with 4 axial power segments, represents a gross approximation of the core axial power shapes. Axial segments 2 and 3, which span the middle elevations in the core region (621.1824 cm to 804.0624 cm), provide a fair approximation of the power shape (i.e., ±10%) around the core mid-plane elevation. Axial segments 1 and 4, which represent the powers in the bottom and top axial segments, respectively, are poor approximations because the average powers in these axial segments do not accurately represent the power gradients at the core boundaries. This is particularly important when evaluating the fluence in the lower regions near elevation H-5 (557.8348 cm). It is estimated that the shroud fluence could be over-estimated by 200-300% at the lower elevations, thus, providing a conservative result at these elevations.

- The axial powers for the fuel assemblies in cycles 13 through 19 assumed that the active fuel height was 381.0 cm (150.0"). The axial powers for these cycles were provided in twenty-five 15.24 cm (6.0") axial segments. Cycles 13 through 19 data is represented with 25 axial power segments and provides the best representation of axial power distribution data that is commonly available from core design methods.

- The core operating data provided for the jet pump riser brace pad benchmark assumed a constant moderator void fraction of about 40% in all regions of the reactor core. A moderator void fraction of 40% is representative of the core average voids around the core mid-plane elevation. Actual moderator coolant densities through the core axial height range from slightly sub-cooled at core entry to about 80% voids at core exit. The moderator void fraction in the peripheral assemblies, which are the primary neutron source for the determination of the shroud fluence, will be slightly lower than the core average due to different flow orificing in the peripheral fuel locations and lower power generation in the peripheral fuel assemblies. Thus, it is expected that the calculated neutron flux will be under-estimated in the upper core elevations due to higher than normal moderator densities and over-estimated in the lower core elevations due to lower than normal water densities.

- The core operating data that was generated for the jet pump riser brace pad benchmark assumed a computer model that is based on traditional discrete ordinate transport methods. These methods employ correlations (approximations) that assume a fission spectrum for the reactor core that is based on core exposure. The RAMA Fluence Methodology employs a more accurate neutron source treatment that uses actual fuel uranium and plutonium number densities in the core region to calculate the fission spectrum and neutron source. In order to perform the jet pump riser brace pad analysis with RAMA, it was necessary to generate fuel isotopic data using a fuel assembly lattice code. The effect of using the calculated fuel isotopics data instead of fuel exposure data to obtain the fission spectrum is not known.

- The reactor operating data provided for the jet pump riser brace pad benchmark omitted plant-specific data for the ex-core regions. Of particular importance is the water densities used in the core reflector, downcomer, and jet pump pipes. Generic water density information was taken from [7] and was used instead of plant-specific water densities. The effect of this assumption cannot be assessed until plant-specific water density inputs are made available.

Axial elevations for the cracks in the Hatch 1 shroud range from about 550.2 cm (about 216.62") at the lower elevation to about 812.5 cm (about 319.88") at the upper elevation. Figure 4 illustrates the axial elevations in a side view of the core shroud.

Cracks at the upper elevation are about 69.5 cm (27.38") above the jet pump brace pad elevation. Because of the assumptions used in the preparation of the Hatch 1 operating data, the results at the upper elevations may be non-conservative relative to the brace pad elevation.

Cracks at the lower elevation are within about 30 cm (12 inches) of the bottom of active fuel. Because of the assumptions used in the preparation of the Hatch 1 operating data, the results at the lower elevations should be more conservative than reported at the brace pad elevations. Note that there is no measurement data to support any statements regarding conservatisms in the model at elevations away from the core mid-plane.

For this analysis, the Hatch 1 shroud spanning the axial height of the core was meshed into 45 azimuthal arcs, 25 axial segments, and 3 radial annuluses, for a total of 3,375 mesh regions. The RAMA computer model is illustrated in Figure 5. Although twenty-five axial segments are illustrated; only segments 1 through 22 were calculated. This provided coverage of all axial crack locations and allowed the computer calculations to be performed within the time allotted for the analysis.

Calculations

The >1.0 MeV and >0.1 MeV neutron fluences were estimated at the vertical and horizontal crack locations in the Hatch 1 shroud through the end of cycle 20. The fluence was estimated using cycle-specific reactor operating data through cycle 19 that is described in [5]. Operating data for cycle 20 was not specifically available. Cycle 20 data was, therefore, approximated using the reactor core power shapes from cycle 19 and the operating history for cycle 20.

Volume-average neutron fluence was calculated for each of the 3,375 mesh regions of the shroud model that is illustrated in Figure 5. Interpolation/extrapolation techniques were used to determine the fluence at the locations of interest, specifically at the inner and outer surfaces of the shroud wall and at the end points of the cracks.

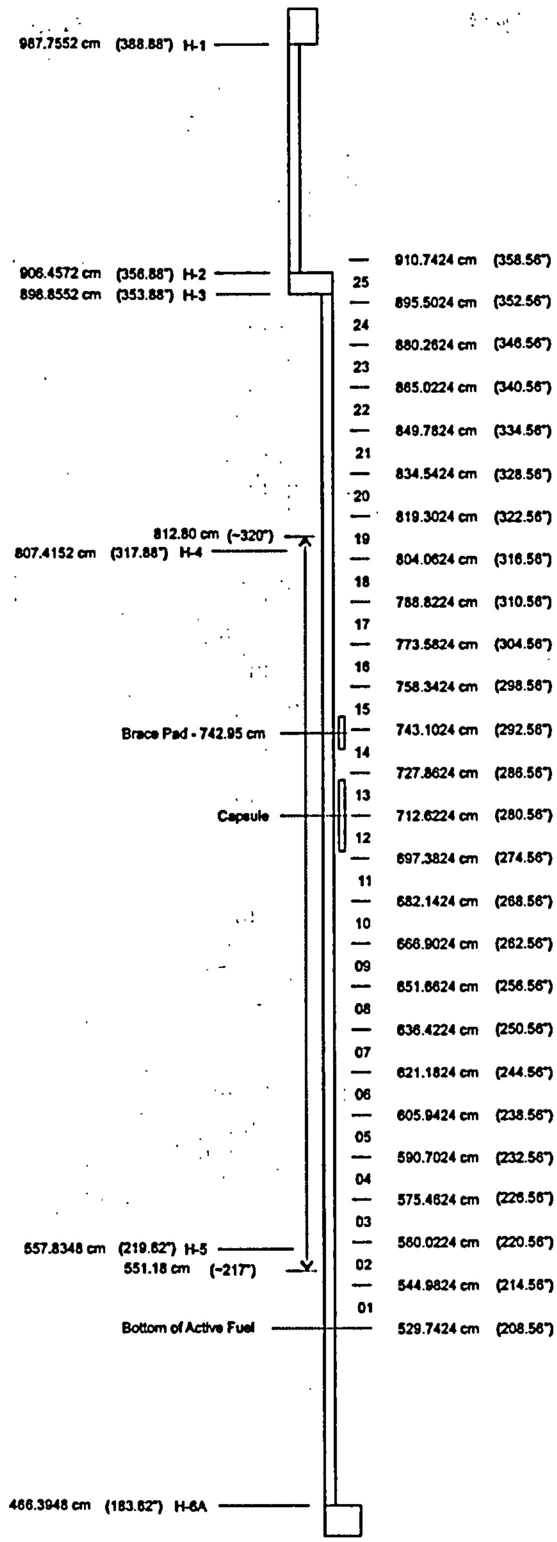


Figure 4
Side View Showing Elevations of the Hatch 1 Shroud

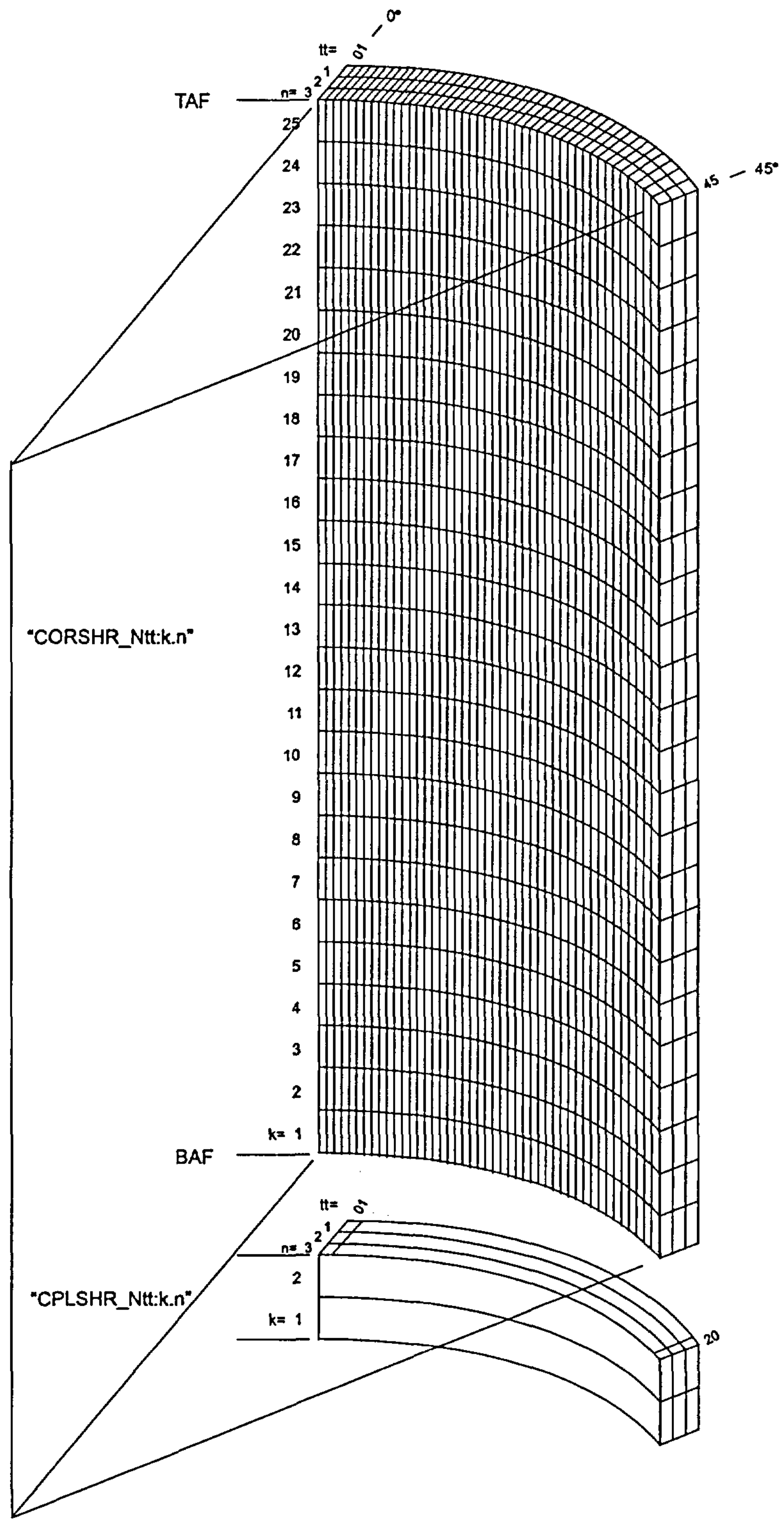


Figure 5
Hatch 1 Shroud Computer Model Mesh Description

- Table 2 lists the >1.0 MeV and >0.1 MeV fluences for the vertical and horizontal cracks identified in Table 1. Fluence values are given for the end points of each of the reported cracks.

Table 2
Fluence Estimates for the Shroud Cracks

Crack ID	Degrees	Elevations (inches) Bottom - Top	>1.0 MeV Fluence Bottom	>1.0 MeV Fluence Top	>0.1 MeV Fluence Bottom	>0.1 MeV Fluence Top
V04-01 Inside 2"	324	317.88 - 319.88	8.95E+20	8.68E+20	1.74E+21	1.68E+21
V05-01 Outside 13.5"	47	284.38 - 297.88	7.61E+20	7.49E+20	1.64E+21	1.61E+21
V05-02 Outside 2"	47	259.88 - 261.88	7.55E+20	7.58E+20	1.62E+21	1.63E+21
V05-03 Outside ¾"	47	255.13 - 255.88	7.42E+20	7.44E+20	1.60E+21	1.60E+21
V05-04 Outside ¼"	47	237.63 - 237.88	6.40E+20	6.42E+20	1.38E+21	1.38E+21
V05-05 Outside 2"	47	233.88 - 235.88	6.20E+20	6.31E+20	1.33E+21	1.36E+21
V05-06 Outside 1.5"	47	226.38 - 227.88	5.75E+20	5.86E+20	1.23E+21	1.26E+21
V05-07 Outside 1.5"	47	218.12 - 219.62	4.77E+20	4.98E+20	1.02E+21	1.07E+21
V05-08 Inside ¾"	53	266.13 - 266.88	**1.23E+21**	**1.23E+21**	2.42E+21	2.42E+21
V05-09 Outside 1.5"	53	218.12 - 219.62	3.92E+20	4.09E+20	8.36E+20	8.73E+20
V06-01 Inside ½"	228	279.38 - 279.88	**1.45E+21**	**1.45E+21**	2.88E+21	2.87E+21
V06-02 Outside 11.5"	233	306.38 - 317.88	5.85E+20	5.21E+20	1.26E+21	1.12E+21
V06-03 Outside 25.5"	232	279.88 - 305.38	6.95E+20	6.64E+20	1.49E+21	1.43E+21
V06-04 Outside 5.25"	232	263.63 - 268.88	6.95E+20	7.00E+20	1.49E+21	1.50E+21
V06-05 Inside 2"	232	219.62 - 221.62	9.35E+20	9.86E+20	1.85E+21	1.95E+21
V08-01 Outside ½"	322	216.62 - 217.12	4.12E+20	4.24E+20	8.78E+20	9.03E+20
H05-01, Inside 10"	228.5	219.62	**1.01E+21**		1.98E+21	
	231.5	219.62	9.66E+20		1.90E+21	

Table 2 shows that vertical cracks V05-08 and V06-01 and horizontal crack H05-01 exceed the fracture toughness threshold of 1.0×10^{21} n/cm^2 for the >1.0 MeV fluence. Each of these cracks are on the inside surface of the shroud wall. Vertical cracks V04-01 and V06-05, which are also on the inside surface, are approaching the toughness threshold with values ranging from 8.68×10^{20} n/cm^2 to 9.86×10^{20} n/cm^2. Several cracks are identified on the outside surface of the shroud wall, however, none of the outer surface cracks show violations of the fracture toughness threshold.

All of the cracks except V05-09 and V08-01 for the >0.1 MeV fluence are shown to have exceeded the fracture toughness threshold. The two exceptions are cracks that are on the outside surface of the shroud wall, although they are approaching the toughness threshold.

The cracks appear to be situated around specific azimuths in the reactor. Reviewing the reactor core configuration shown in Figure 3, it is observed that the cracks are nearly aligned with the fuel bundles that are closest to the shroud wall. This should be expected since these locations are closest to the neutron source.

Vertical Cracks

Figures 6 through 11 contain axial plots of the >1.0 MeV and >0.1 MeV neutron fluences for the vertical welds in the Hatch 1 shroud.

Figure 6 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical cracks at azimuth 53 degrees (near V-5) between the elevations H-4 and H-5. Both the inside and outside wall fluences are shown.

Figure 7 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical crack at azimuth 47 degrees (near V-5) between the elevations H-4 and H-5. Only the outside wall fluence is shown.

Figure 8 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical cracks at azimuths 228 and 233 degrees (near V-6) between the elevations H-4 and H-5. The inside wall fluence is given for the 228 azimuth position and the outside wall fluence is given for the 233 azimuth position.

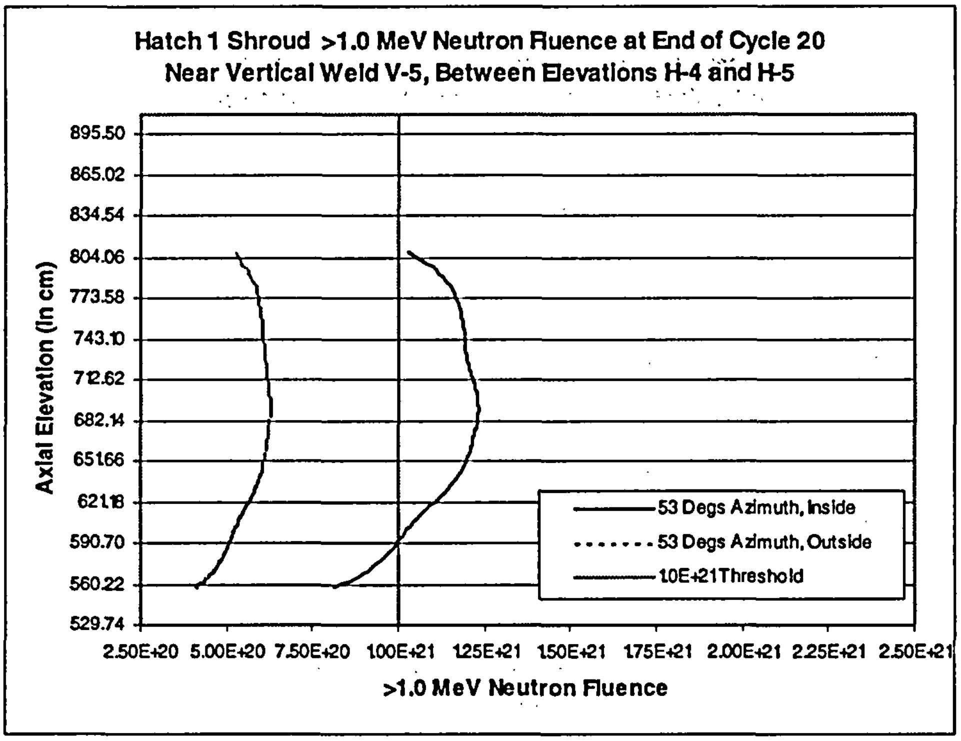
Figure 9 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical cracks at azimuth 232 degrees (near V-6) between the elevations H-4 and H-5. Both the inside and outside wall fluences are shown.

Figure 10 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical cracks at azimuth 322 degrees (near V-8) between the elevations bottom of active fuel and H-5. Only the outside wall fluence is shown.

Figure 11 shows the axial >1.0 MeV and >0.1 MeV neutron fluences for the vertical cracks at azimuth 324 degrees (near V-4) between the elevations H-4 and top of active fuel. Only the inside wall fluence is shown.

Horizontal Cracks

Figure 12 shows azimuthal plots of the >1.0 MeV and >0.1 MeV neutron fluences for the H-5 weld . Azimuthal fluences are shown for both the inside and outside wall surfaces.



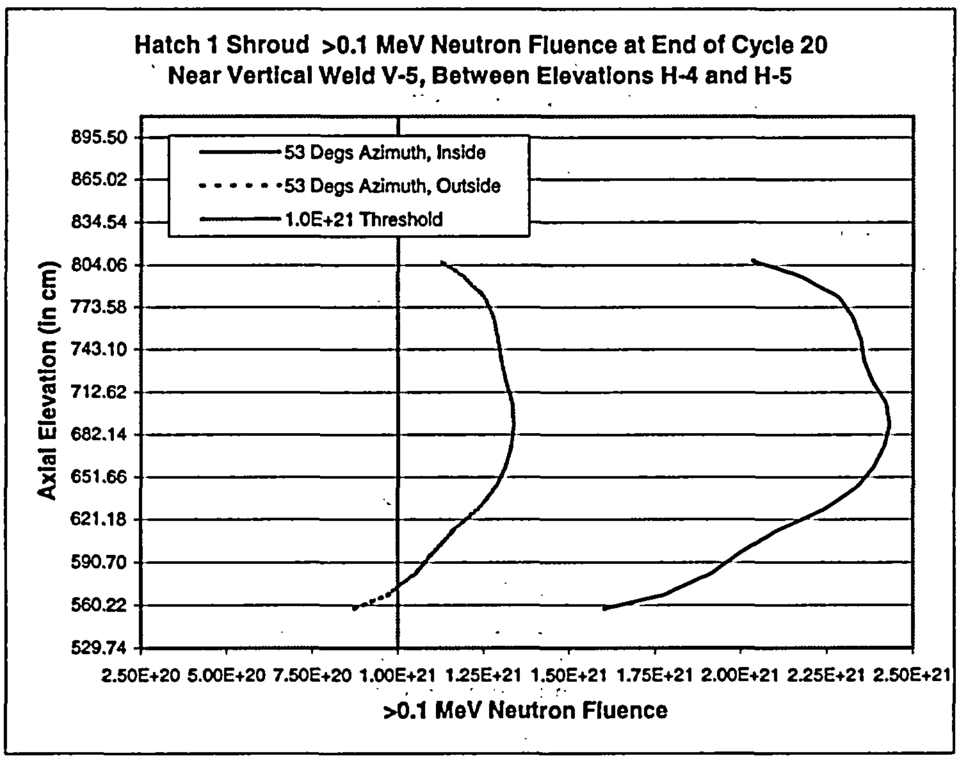
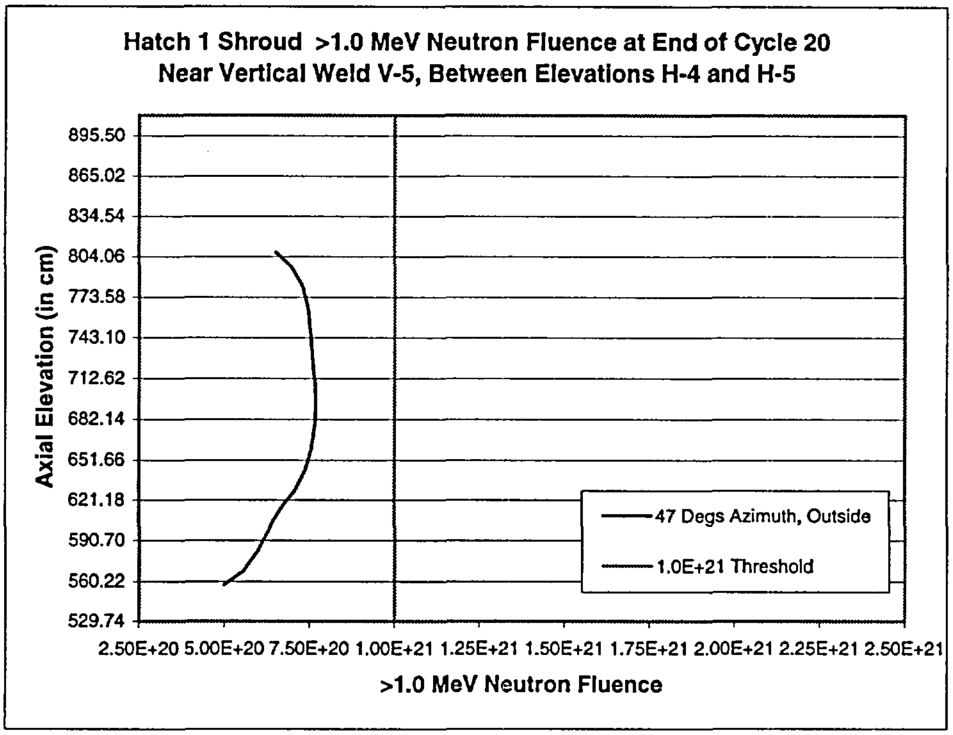


Figure 6
Hatch 1 Shroud Axial Neutron Fluence at Azimuth 53 Degrees



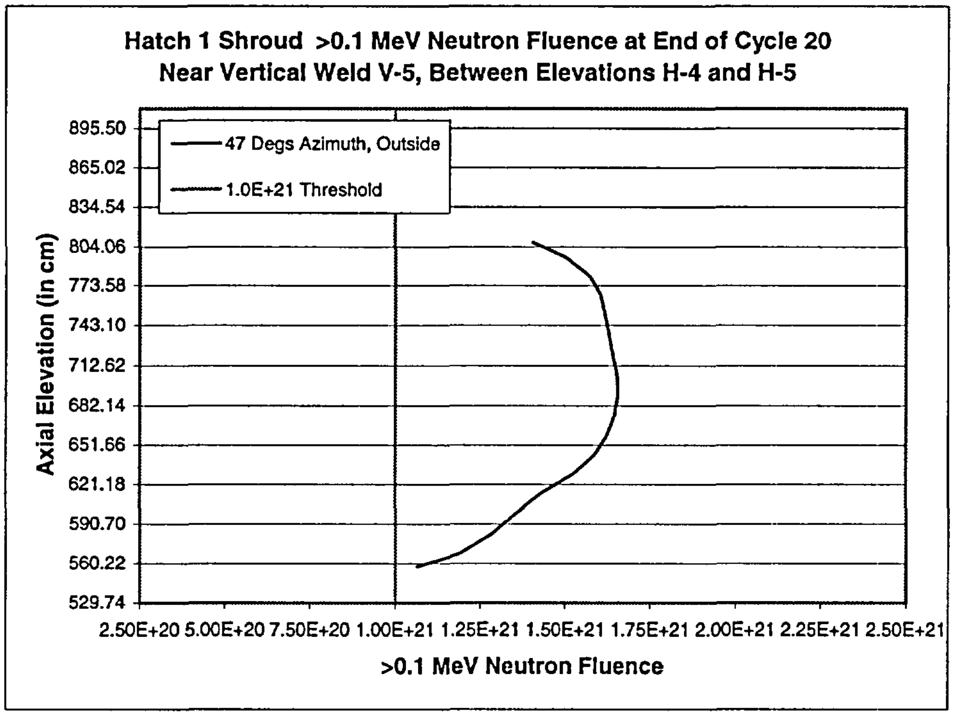
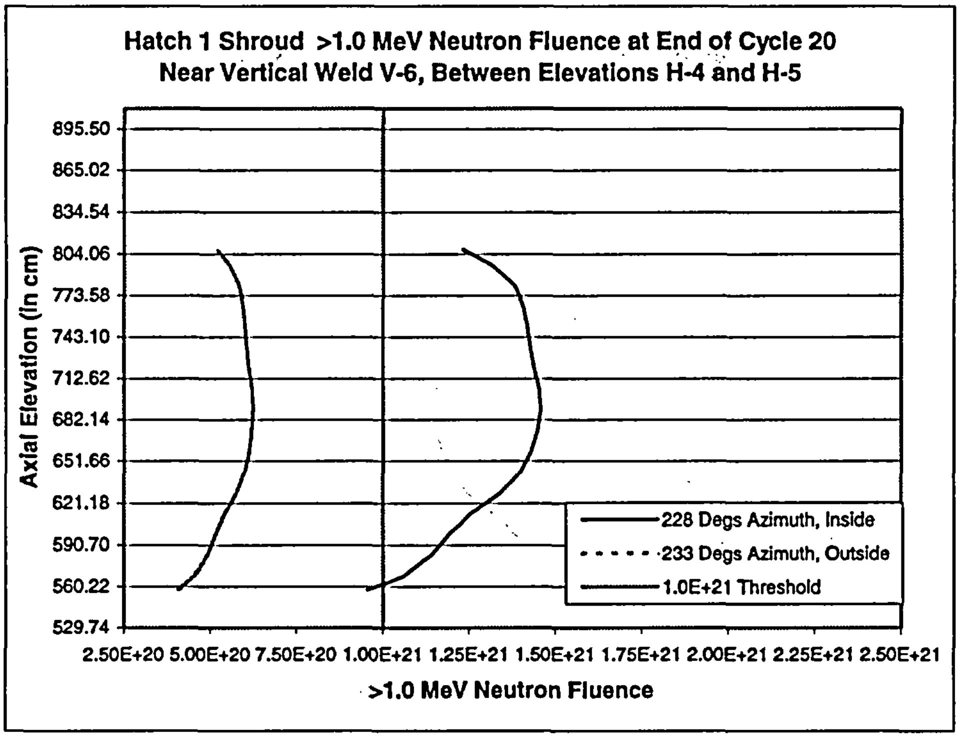


Figure 7
Hatch 1 Shroud Axial Neutron Fluence at 47 Degrees



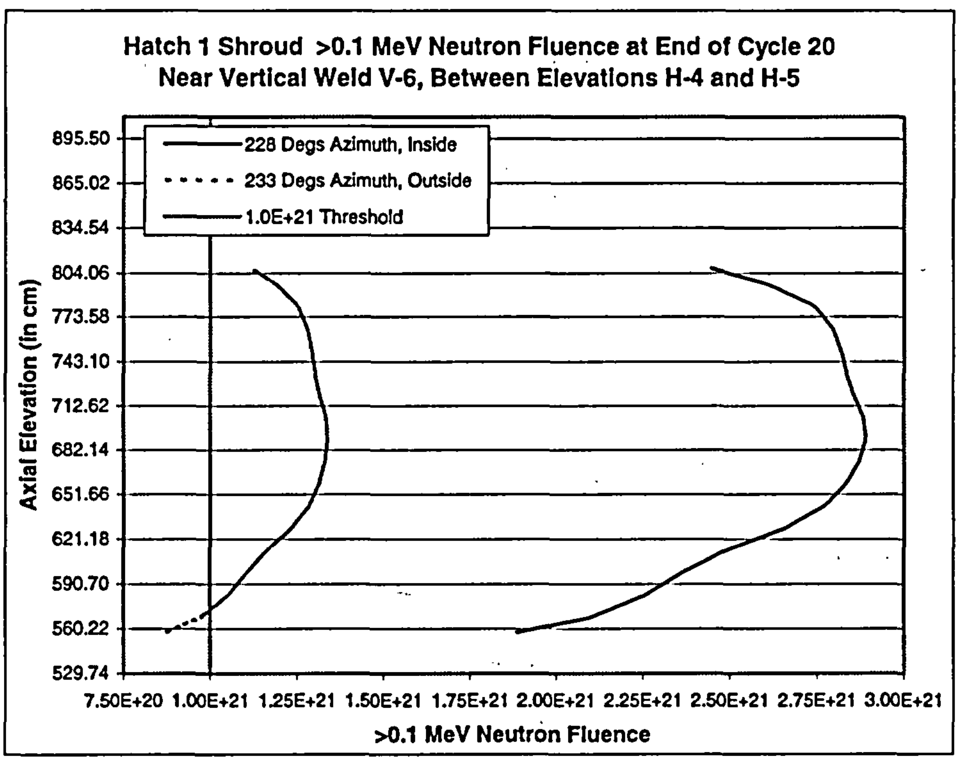
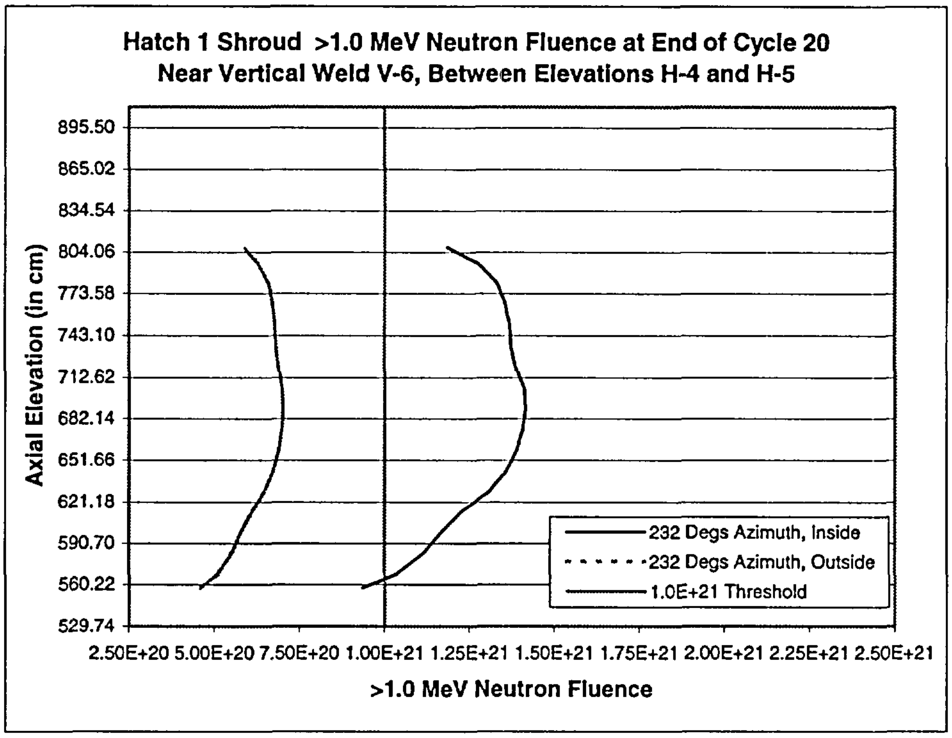


Figure 8
Hatch 1 Shroud Axial Neutron Fluence at 228 and 233 Degrees



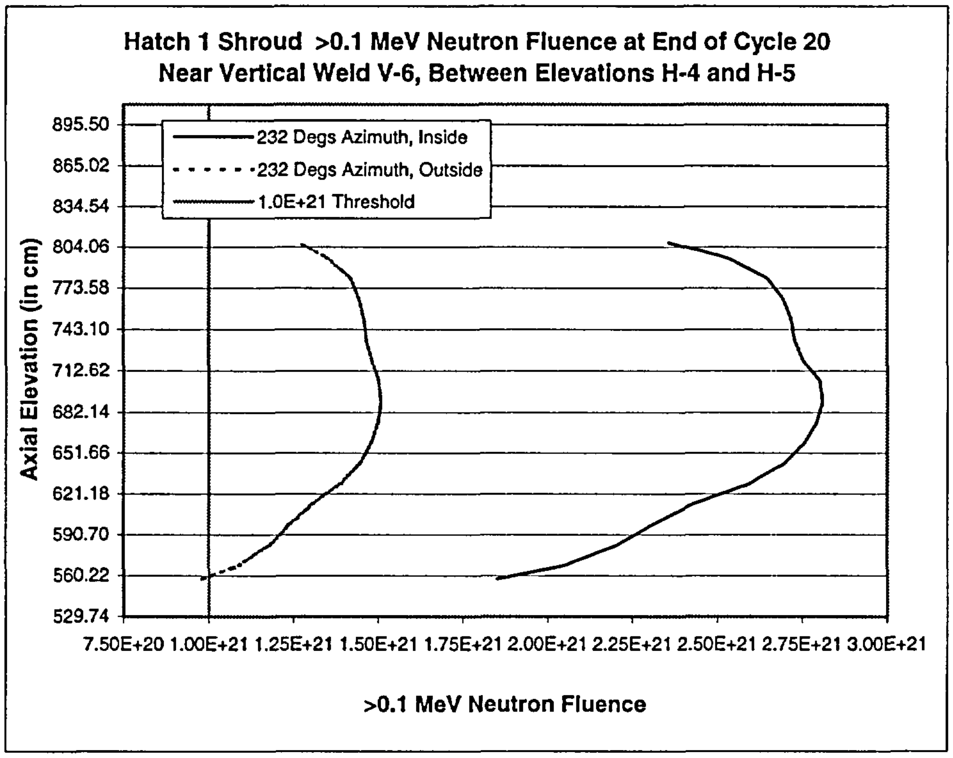
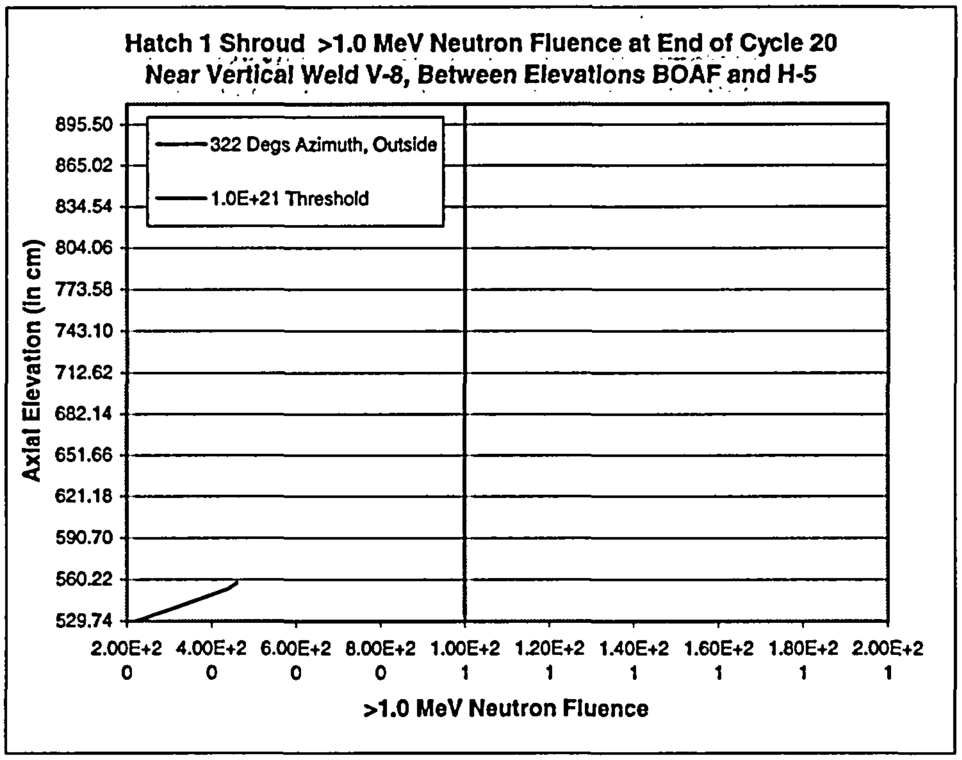


Figure 9
Hatch 1 Shroud Axial Neutron Fluence at 232 Degrees



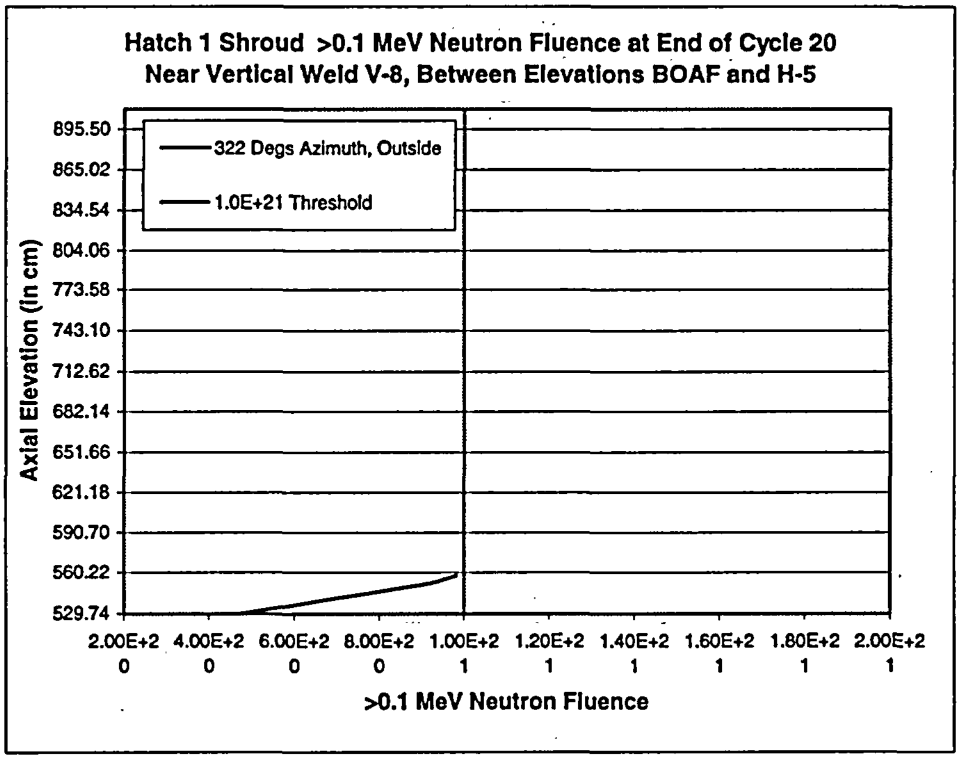
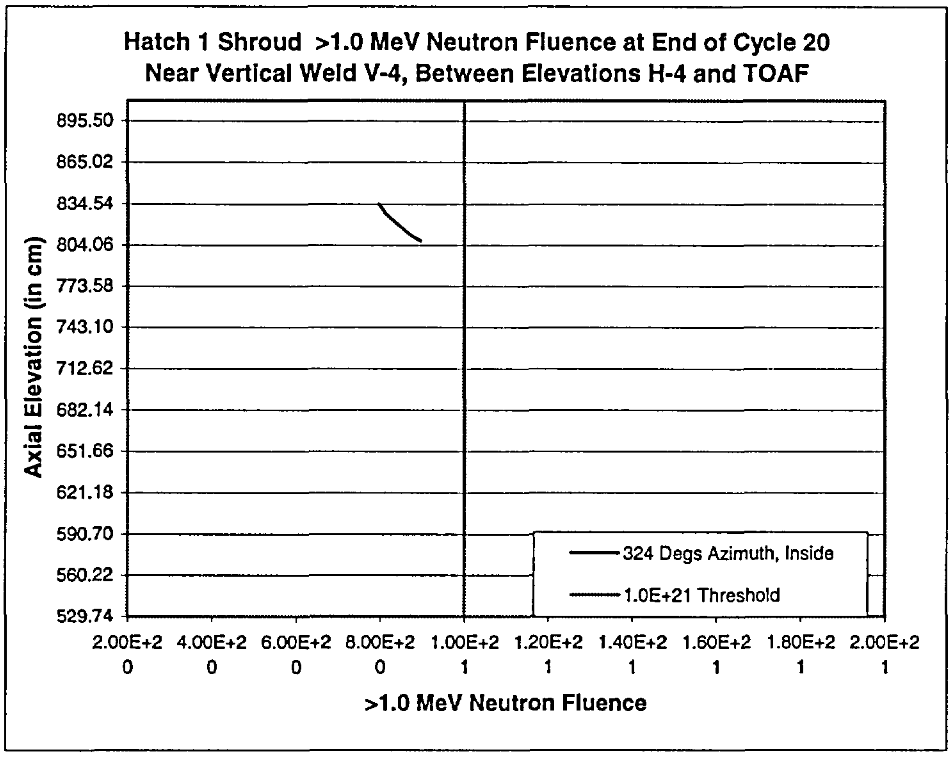


Figure 10
Hatch 1 Shroud Axial Neutron Fluence at 322 Degrees



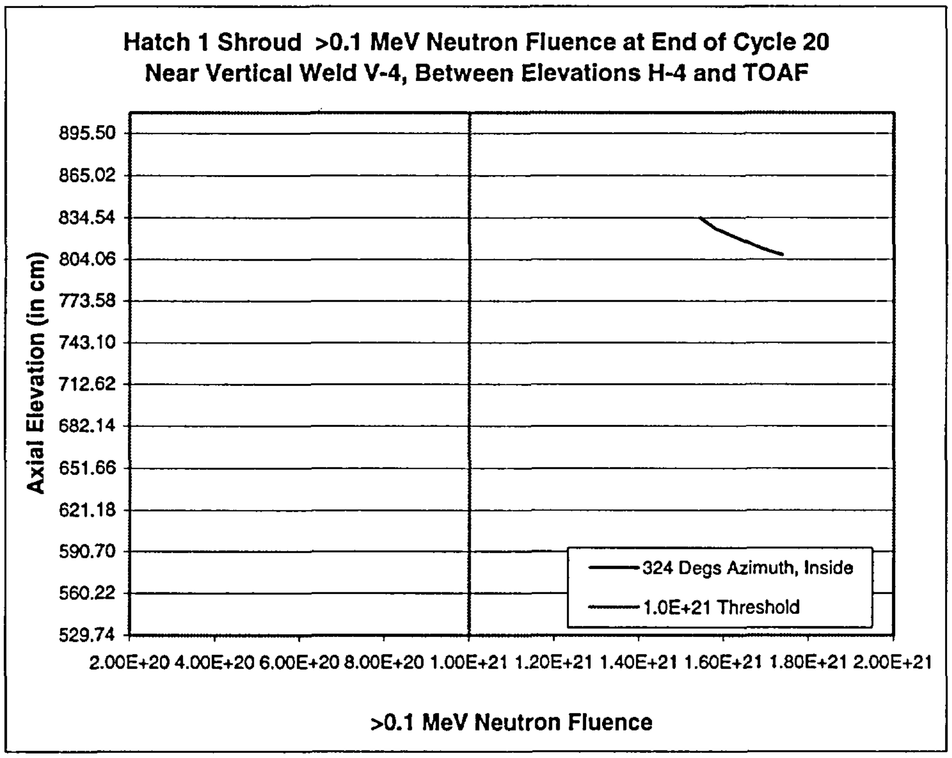
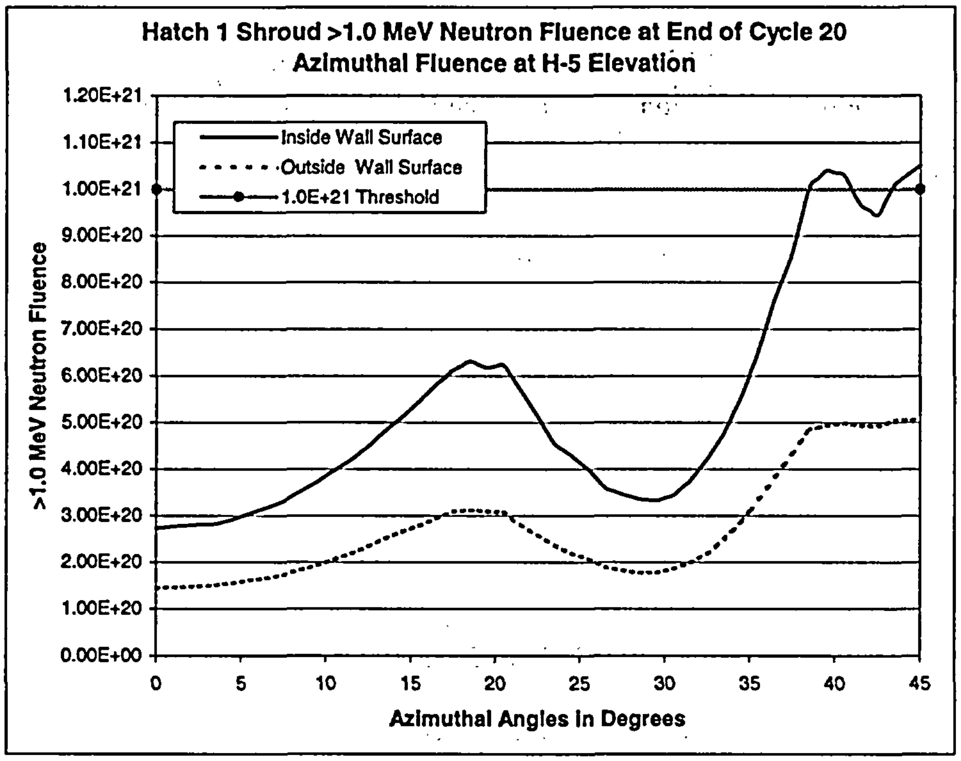


Figure 11
Hatch 1 Shroud Axial Neutron Fluence at 324 Degrees



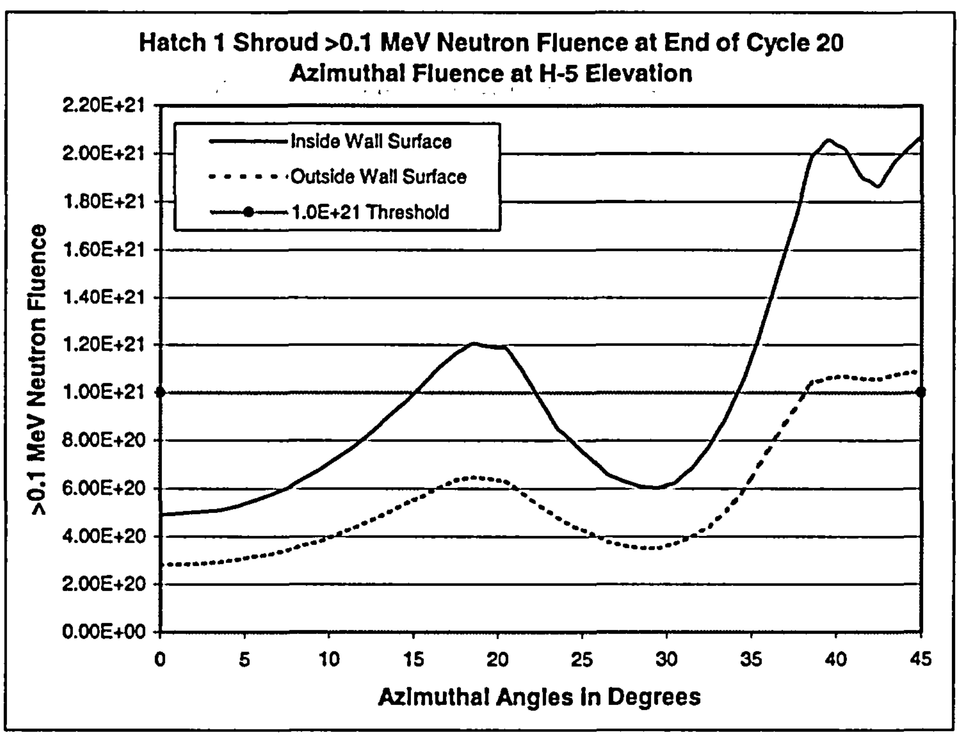


Figure 12
Hatch 1 Shroud Azimuthal Neutron Fluence at H-5 Elevation

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