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171-9001

September 9, 2002

E&L-039-02

Mr. E. William Brach, Director
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards, NMSS
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Brach:

CONSOLIDATED SAFETY ANALYSIS REPORT FOR THE IF-300, CERTIFICATE OF COMPLIANCE 9001

Enclosed is our application for the revision to the Consolidated Safety Analysis Report (CSAR) for the IF-300, Certificate of Compliance No. 9001. We request that you approve the revision to the CSAR and revise the Certificate for IF-300 to reflect the following changes.

1. The maximum initial uranium content per Group III PWR assembly has been increased from 437 to 442 kg. The additional analyses demonstrating that this fuel type with additional uranium content meets regulatory requirements are contained in Appendix 9.3 to Appendix D, which is included in Volume IV of the CSAR. Appendix 9.3 is provided as a non-proprietary document.
2. Several pages in Volume IV Appendix D of the CSAR have been revised to reflect the addition of Appendix 9.3. Page D-3-1 contains proprietary information. Non-proprietary and proprietary versions of page D-3-1 are provided.

There are several attachments to this letter, including the CSAR revision pages themselves. The following is a discussion of the content and purpose of each attachment.

Attachment 1: A Revision Control sheet to the CSAR showing which pages have been changed from the current revision.

Attachment 2: The replacement pages for Volume IV. (Non-proprietary)

Attachment 3: Appendix 9.3 of Appendix D, Volume IV (Non-proprietary)

Attachment 4: Page D-3-1 (Proprietary)

NMSS01
PROP

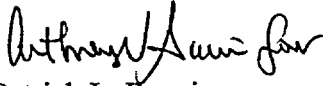
Mr. E. William Brach
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We request completion of action on this request by December 2, 2002 to allow development of appropriate procedures and controls for use of the cask with the higher uranium content Group III PWR fuel from the H. B. Robinson Plant. The owner of the cask, CP&L, needs to ship Group III fuel from the H. B. Robinson Plant in early 2003 to maintain a prudent operating margin. This need stems from the U.S. Department of Energy's failure to begin taking spent fuel in 1998 as required by the Nuclear Waste Policy Act of 1982, as amended, and by its contracts with the nuclear utilities.

Should you or members of your staff have any questions about the application, please feel free to contact Mark Whittaker at (803) 758-1898.

Sincerely,



Patrick L. Paquin
General Manager – Engineering and Licensing

Attachments: As stated

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ATTACHMENT 1

REVISION CONTROL SHEET

REVISION CONTROL SHEET

TITLE: Consolidated Safety
 Analysis Report for IF-300
 Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
1-i	3	Last revision prepared by General Electric Company.
1-1 & 1-2	"	Incorporates all C of C 9001, Revision 29 references
2-i & 2-ii	"	From 2/8/84 through 5/10/85.
2-1 - 2-15	"	A vertical line on the right hand margin indicates a
3-i & 3-ii	"	Revision. "N" denotes new information while "E"
3-1 - 3-16	"	Denotes an editorial change.
4-i - 4-ii	"	
4-1 - 4-21	"	
5-i - 5-vi	"	
5-1 - 5-311	"	
6-i - 6-iv	"	
6-1 - 6-82	"	
7-i - 7-ii	"	
7-1 - 7-22	"	
8-i & 8-ii	"	
8-1 - 8-22	"	
9-i & 9-ii	"	
9-1 - 9-6	"	
10-i & 10-ii	"	
10-1 - 10-15	"	
A-i & A-ii	"	
V1-i - V1-iv	"	
V1-1 - V1-52	"	
V1-A-i/ii	"	
V1-A-1 -	"	
V1-A-3	"	
V1-B-i/ii	"	
V1-B-1 &	"	
V1-B-2	"	
V1-C-i/ii	"	
V1-C-1 -	"	
V1-C-8	"	

REVISION CONTROL SHEET

TITLE: Consolidated Safety
Analysis Report for IF-300
Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
V1-D-i -	3	
V1-D-vi	"	
V1-D-1 -	"	
V1-D-132	"	
V1-E-i &		
V1-E-ii	"	
V1-E-1 -	"	
V1-E-34	"	
V2-i - V2-iv	"	
V2-1 - V2-64	"	
V3-i - V3-iv	"	
V3-1 - V3-32	"	
VI-i & VI-ii	"	
VI-1 - VI-6	"	
i - viii	4	First revision prepared by VECTRA Technologies, Inc.
1-1	"	Incorporates C of C 9001, Revision 29 references
2-1 & 2-2	"	From 7/26/90 through 4/25/99.
2-3 & 2-3a	"	A vertical line on the left hand margin indicates a
2-4 & 2-5	"	Revision.
2-8	"	
2-10 & 2-11	"	
2-14 & 2-15	"	
3-1 & 3-1a	"	
3-2a	"	
3-16	"	
4-1, 4-3, 4-5	"	
4-9 & 4-10	"	
5-1, 5-3, &	"	
5-5	"	
5-101 & 5-270	"	

REVISION CONTROL SHEET

TITLE: Consolidated Safety
 Analysis Report for IF-300
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DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
6-1 & 6-1a	4	
6-2 & 6-2a	"	
6-33 & 6-35	"	
7-1 & 7-2	"	
8-1 & 8-3	"	
8-19 & 8-20	"	
9-1, 9-2, 9-6	"	
10-4	"	
A-i/ii	"	
A-iii -	"	
A-xiii	"	
A-1-1 -	"	
A-1-10	"	
A-2-1 -	"	
A-2-338	"	
A-3-1 -	"	
A-3-86	"	
A-4-1 - A-4-5	"	
A-5-1 -	"	
A-5-92	"	
A-6-1 -	"	
A-6-131	"	
A-7-1 & A-8-1	"	
A-9-1 -	"	
A-9-13	"	
B-i/ii	"	
B-iii & B-iv	"	
B-1 - B-55	"	
C-i/ii	"	
C-iii/iv	"	
C-1 - C-14	"	

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TITLE: Consolidated Safety
 Analysis Report for IF-300
 Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
1-1 - 1-3	5	First revision prepared by CNS, Inc.
2-1 - 2-3a	"	Incorporates C of C 9001, Revision 34 references
2-10 - 2-12a	"	From 01/09/1998 through 11/09/2000.
2-14 & 2-15	"	A vertical line on the left hand margin indicates
3-1 & 3-1a	"	changed text. The date in the header reflects the
4-ii & 4-22	"	date of the submittal.
5-3 & 5-4	"	
6-1 - 6-2	"	
8-1 & 8-2	"	
9-1 -& 9-2	"	
9-5 & 9-6	"	
10-i - 10-ii	"	
10-1 - 10-15	"	
D-i - D-vii	"	
D-1-1 - D-1-2	"	
D-2-1 - D-2-3	"	
D-3-1- D-3-15	"	
D-4-1- D-4-16	"	
D-5-1- D-5-13	"	
D-6-1 -	"	
D-6-170	"	
D-7-1- D-7-11	"	
D-8-1 - D-8-3	"	
D-9-1- D-9-46	"	

REVISION CONTROL SHEET

TITLE: Consolidated Safety
Analysis Report for IF-300
Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	HEADER DATE	REMARKS
1-1 - 1-2	5	August 2001	First revision prepared by Duratek, Inc. The document revision level is not changed since only a fraction of the SAR is changed. A vertical line in the margin indicates changed text. The date in the page header reflects the date of the change. The revision status of a particular page is noted by the revision number and header date.
2-10 - 2-11	"	"	
4-13	"	"	
6-79 - 6-80	"	"	
9-5 - 9-6	"	"	
10-i - 10-ii	"	"	
10-1 - 10-15	"	"	
E-i - E-v	"	January 2002	
E-1-1	"	"	
E-2-1 - E-2-3	"	"	
E-3-1 - E-3-9	"	"	
E-4-1 - E-4-26	"	"	
E-5-1 - E-5-34	"	"	
E-6-1 - E-6-3	"	"	
E-7-1 - E-7-2	"	"	
E-8-1 - E-8-453	"	"	
D-1-2 - D-1-2a	5	September 2002	Second revision prepared by Duratek, Inc. The document revision level is not changed since only a fraction of the SAR is changed. A vertical line in the margin indicates changed text. The date in the page header reflects the date of the change. The revision status of a particular page is noted by the revision number and header date.
D-2-1 - D-2-1a	"	"	
D-3-1	"	"	
D-4-1	"	"	
D-5-1 - D-5-1a	"	"	
D-6-5 - D-6-5a	"	"	
D-7-2 - D-7-2a	"	"	
D-9-47 - D-9-80	"	"	

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ATTACHMENT 2

REPLACEMENT PAGES
VOLUME IV

Appendix D-9.3 was added to address GIII PWR fuel assemblies with a maximum of 442 kg of uranium per assembly because it was discovered that some of the GIII PWR fuel had slightly greater uranium masses than the 437 kg originally analyzed. Appendix D-9.3 addresses the impact of the slight increase in uranium mass on the neutron and gamma source strengths, decay heat, releasable source terms during hypothetical accident conditions, and criticality safety of the IF-300 Package. It is concluded that all of the transportation requirements specified in 10 CFR 71, as well as the limits specified in the IF-300 C of C (C of C No. 9001) and the Consolidated Safety Analysis Report (CSAR 1995) are satisfied when shipping GIII PWR fuel assemblies with uranium masses up to 442 kg/assembly as long as the other loading restrictions listed in Table D-1.1 are satisfied.

Table D-1.1

Characteristics of Group III Assemblies

Parameter	PWR Fuel	BWR Fuel
Fuel Type	15x15 SPC PWR for Westinghouse Class 15x15 Reactors	GE-7, 8, 9, 10 & 13 BWR Fuel for General Electric BWR/4 Plant Design
Uranium Weight	437 442 kg/assembly ^a	187 kg/assembly
Number of Assemblies	6 ^a	17 channeled
Maximum Assembly Average Burnup	45 GWD/MTU	45 GWD/MTU
Maximum Lattice Average Enrichment ^b	4.25 wt% ²³⁵ U	4.25 wt% ²³⁵ U
Minimum Blanket Length ^c	6 inches	6 inches
Minimum Cooling Time	5 years	4 years

^a The center location in the PWR basket must be left empty. If an assembly is loaded in the center PWR basket location and a peripheral location is instead left empty, the cask will not meet the criticality safety criteria with 6 assemblies at 4.25 wt% ²³⁵U.

^b The maximum lattice average enrichment is defined as the maximum planar average enrichment of any axial plane from the bottom to the top of the fuel assembly. Although some individual fuel rod enrichments may exceed this enrichment, the average enrichment for

- every axial slice across the fuel assembly must not exceed the maximum lattice average enrichment.
- c The minimum length of natural UO₂ fuel above and below the enriched portion of the active fuel.
 - c Appendix D-9.3 contains the evaluation for an increase in the uranium mass from 437 kg/assembly to 442 kg/assembly.

A plug was designed for the PWR fuel basket to prevent inadvertent loading of the center basket location. The plug has an outer lip that rests on the top of the basket when the plug is inserted into the basket. The plug extends above the basket 3.625 in and down into the basket approximately 8 in to prevent it from accidentally coming out during fuel loading or transport. The bottom of the plug is approximately 7 in above the top of the active fuel and the plug itself adds only a small amount of steel in a region of low importance from a criticality standpoint. Also, because the plug is located above the empty center basket location, the reactivity effect due to reflection of neutrons back into the fuel is negligible.

D-2.0 ANALYSIS METHODOLOGY

The approach used in this appendix to address the effect of the change on the pre-1995 IF-300 certification is described in this section. Appendix D-9.3 contains the evaluation for an increase in the uranium mass from 437 kg/assembly to 442 kg/assembly for the GIII PWR fuel. Appendix D-9.3 addresses the safety implications of the increased uranium mass for each of the assessments described in Sections 2.1 through 2.5.

D-2.1 Structural Assessment

The requested change deals with increasing the permissible burnup and enrichment of fuel to be shipped in the IF-300 package for specific types of PWR and BWR fuel, which are designated as Group III fuel assemblies. The Group III fuel assemblies have outer dimension envelopes that are compatible with the Group I and II fuel designs, and fuel assembly weights that are bounded by the Group I and II fuel designs already permitted under the current IF-300 Cask C of C (C of C No. 9001). The total weight of a PWR Group III fuel assembly is 653.2 kg, and the maximum assembly weight for the BWR Group III fuel designs is 301.2 kg (GE-7). This results in a total fuel assembly loading of 3,920 kg (6 x 653.2 kg) for the PWR cask and 5,120 kg (17 x 301.2 kg) for the BWR cask. This is below the structural evaluation design basis assembly loading of 4,920 kg for the PWR cask and 5,205 kg for the BWR Cask with the channeled fuel assembly basket. The PWR cask analysis basis fuel loading of 4,920 kg was obtained by subtracting the PWR basket weight (2,050 kg from p. 5-100) from the maximum PWR basket weight with fuel (6,970 kg from p. 5-99). The BWR cask analysis basis fuel loading of 5,205 kg was obtained from Table A-2.2-1 (p. A-2-8) for the channeled BWR fuel basket. Therefore, there is no change in the structural aspects of the contents to be shipped.

Furthermore, since Section 4.0 indicates that the number of moles of residual gas available for release in the cask cavity and the total decay heat load will remain within the pre-1994 licensed limit of 0.5 moles and 40,000 Btu/h, respectively, no change in the internal pressure will occur. Therefore, the proposed change has no effect on the structural analysis of the cask upon which the current C of C (C of C No. 9001) is based.

D-3.0 NEUTRON AND GAMMA SOURCE STRENGTH AND DECAY HEAT CALCULATION

This section determines the neutron and gamma source strengths and the decay heat for the higher burnup and higher enrichment fuel assemblies described in Table D-1.1. The calculation of the neutron and gamma source strengths are performed using ORIGEN2 (Savino 1999). ORIGEN2 is a computer code which calculates the buildup and decay of radioactive materials. ORIGEN2 code was developed by the Oak Ridge National Laboratory and is distributed through the Radiation Shielding Information Computational Center.

Sections D-3.1 and D-3.2 describe the ORIGEN2 models for the GIII PWR and BWR fuel assemblies, respectively. Sections D-3.3 and D-3.4 describe the results of the ORIGEN2 calculations for the GIII PWR and BWR fuel assemblies, respectively. Sections D-3.1 and D-3.3 are based on PWR fuel with a uranium mass of 437 kg/assembly. Appendix D-9.3 contains ORIGEN2 models for the GIII PWR fuel with a uranium mass of 442 kg/assembly.

D-3.1 ORIGEN2 Input File Preparation for Group III PWR Fuel

The GIII PWR fuel assembly is assumed to consist of four axial regions for the source term calculation: the active fuel region, the gas plenum region, the upper tie plate region, and the lower tie plate region.

The GIII PWR fuel assembly consists of 204 fueled rods, 20 control rod guide tubes, and one instrument tube. The fuel rod cladding material is Zircaloy-4. The cladding extends into the top plenum zone and the bottom end fitting zone. The total rod length is [REDACTED] in out of which 144 in is in the active fuel zone, [REDACTED] in is in the top plenum zone, and [REDACTED] in is in the lower tie plate zone. The Zircaloy-4 weight due to cladding in various regions of the fuel assembly is assumed to be proportional to the cladding lengths in these regions. The component weights were taken from Kunita (1998) and are described below.

The highlighted data is proprietary information to Siemens Power Corporation

D-4.0 DETERMINATION OF ACTIVATION PRODUCTS AND CONTAINMENT EVALUATION

The generation of the activation product and containment evaluation source terms is included in this section. Sections D-4.1 and D-4.2 contain the results for the GIII PWR fuel and the GIII BWR fuel, respectively. Appendix D-9.3 contains the results for GIII PWR fuel with a uranium mass of 442 kg/assembly.

D-4.1 Group III PWR Fuel

D-4.1.1 Upper Nozzle Hardware Activation Product Source

Activation analysis is performed to determine the impact of higher burnup on the ^{60}Co content of the upper nozzle of the GIII PWR fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years. This evaluation conservatively uses a maximum burnup of 50 GWD/MTU for the PWR fuel, which bounds the results for 45 GWD/MTU.

An activation analysis based on a unit mass (one kg) of Cobalt for a burnup of 35 GWD/MTU is contained in Appendix D-9.1, ORIGEN2 run CO265US. Also the activation analysis based on a unit mass (one kg) of Cobalt for a burnup of 50 GWD/MTU is contained in Appendix D-9.1, ORIGEN2 run CO345UE. A comparison of the results is included in Section D-7.

D-4.1.2 Containment Evaluation Source Terms

The increase in the fuel assembly enrichment limit from 4 wt% ^{235}U to 4.25 wt% ^{235}U will impact the releasable source terms from the IF-300 Package. NUREG/CR-6487 (NRC 1996) identifies 3 sources which should be considered when determining the releasable source term for packages designed to transport irradiated fuel rods: 1) crud spallation from fuel rods, 2) release of fines from cladding breaches, and 3) source activity from gases and volatiles released due to cladding breaches. Each of these sources is addressed below. The fission gas product moles calculation is included in Section D-4.1.2.4.

D-5.0 COMPARISON WITH IF-300 PACKAGE C OF C

The key evaluation parameters affected by the increased burnup and enrichment are:

- Decay Heat
- Gamma Shielding
- Neutron Shielding
- Hardware Activation
- Containment Evaluation Source Terms
- Fission Gas Product Moles Source Term
- Criticality

Section D-5.1 contains a comparison of the key evaluation parameters for the existing analysis-basis and those for the IF-300 cask loaded with six higher enrichment GIII PWR fuel assemblies from Robinson. Note that the evaluation parameters for the PWR fuel were generated for a burnup of 50 GWD/MTU and a 5 year cooling time, which bounds the values for 45 GWD/MTU and a 5 year cooling time. Appendix D-9.3 contains the comparison for the GIII PWR fuel with a uranium mass of 442 kg/assembly. Section D-5.2 contains a comparison of the key evaluation parameters for the existing analysis-basis and those for the IF-300 cask loaded with seventeen higher burnup and enrichment GIII BWR fuel assemblies from Brunswick. The criticality safety aspects of the evaluation are contained in Section 6.0.

D-5.1 Results for Group III PWR Fuel

D-5.1.1 Decay Heat

The IF-300 shipping cask C of C limit is 40,000 Btu/h, which corresponds to 5,714 Btu/h/FA for the analysis-basis fuel assembly heat loading for the cask loaded with seven assemblies. The maximum decay heat from the Group III PWR fuel assemblies with a burnup of 50 GWD/MTU, a minimum enrichment of 3.45 wt% ²³⁵U, and a cooling time of 5 years is 4,768 Btu/h per assembly, or a total of 28,608 Btu/h for a cask loading of six fuel assemblies. Therefore, the total cask heat loading and the fuel assembly heat loading for SPC PWR fuel with a burnup of 50 GWD/MTU and a cooling time of

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5 years are well within the IF-300 shipping cask
analysis-basis values. The uranium inventory of the

D-6.1.2 Summary of Criticality Evaluation Results

Table D-6.1-2 contains a summary of the key criticality evaluation results for the IF-300 Cask. Appendix D-9.3 addresses the impact on the criticality evaluation of an increase in the uranium loading from 437 to 442 kg/assembly for GIII PWR fuel.

Table D-6.1-2

Summary Table of Criticality Evaluations

	6 Group III PWR Fuel Assemblies with 4.25 wt% ²³⁵ U	17 Group III BWR Fuel Assemblies with 4.25 wt% ²³⁵ U
Normal Conditions of Transport (NCT)		
Number of Undamaged Packages	∞	∞
Single Unit Maximum k_{tot}^a	0.34574	0.34566
Infinite Array Maximum k_{tot}^a	0.38123	0.37208
k_{eff} Limit	0.95	0.95
Hypothetical Accident Conditions (HAC)		
Single Unit Maximum k_{tot}^a	0.94427	0.92846
Infinite Array Maximum k_{tot}^a	0.94636	0.93625
k_{eff} Limit	0.95	0.95

^a $k_{tot} = k_{eff}$ including bias and uncertainties (see Sections D-6.4-3 and D-6.7).

D-6.1.3 Transport Index

The maximum calculated k_{tot} is less than 0.95 for infinite arrays of IF-300 casks under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) when the casks are loaded with six 4.25 wt% ²³⁵U GIII PWR fuel assemblies with the center basket location empty, or with seventeen 4.25 wt% ²³⁵U GIII BWR fuel assemblies. Therefore, a criticality Transport Index of zero (0) is justified.

D-6.2 Spent Nuclear Fuel Contents

Prior to 1991, the IF-300 Cask was licensed for transport of up to seven PWR or eighteen unchanneled BWR irradiated fuel assemblies. The enrichment limit was 4 wt% ^{235}U for both the PWR and BWR fuel. The

Appendix D-9.3 contains the evaluation for an increase in the uranium loading for the GIII PWR fuel assemblies from 437 to 442 kg/assembly. It was discovered that some of the GIII PWR fuel had slightly greater uranium masses than the 437 kg originally analyzed.

Table D-7.1-1

Comparison of Key Evaluation Parameters for Group III 15x15 PWR Fuel With an Exposure of 50 GWD/MTU^a and a 5-Year Cooling Time and the Analysis-Basis Values

PARAMETER ^b	ANALYSIS BASIS	SIX GROUP III PWR FAS	MARGIN, %
NEUTRON SOURCE (NEUTRON/S/CASK)	1.87E+09	4.08E+09	-218
GAMMA SOURCE (GAMMA/S/CASK)	7.90E+16	5.54E+16	30
DECAY HEAT (BTU/H/CASK)	40,000	28,608	28
RELEASABLE ⁸⁵ KR (Ci/CASK)	13,086	8,904	32
TOTAL RELEASABLE RESIDUAL FISSION PRODUCT MOLES	0.5	0.32	36
	(Ci OF ⁶⁰ CO/KG COBALT)	(Ci OF ⁶⁰ CO/KG COBALT)	
HARDWARE ACTIVATION	1.42E+04	1.33E+04	6

^a This evaluation conservatively uses a maximum burnup of 50 GWD/MTU for the PWR fuel, which bounds the results for 45 GWD/MTU.

^b Section D-5.1 addresses additional containment evaluation parameters (i.e., crud spallation, total gases and volatiles, and releases from fuel fines) and qualitatively demonstrates that the GIII PWR fuel is bounded by the analysis basis PWR fuel for these parameters.

D-7.1-1 Estimates of NCT and HAC Dose Rates for the IF-300 PWR Cask

NCT Dose Rates at the IF-300 PWR Cask Surface

The neutron and gamma dose rates at the surface of the IF-300 cask are estimated using the method described in Appendix B. This method involves calculating the neutron and gamma dose rates by scaling the dose rates for the analysis basis BWR fuel loading by the ratio of the neutron (or gamma) source terms for the PWR fuel to the neutron (or gamma) source terms for the analysis basis BWR fuel. The dose rates used are from the 17 element channeled BWR fuel basket (CSAR 1995) shielding evaluation (Table A-5.4-5, p. A-5-35) since

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the CSAR does not report the total neutron and gamma dose rates with dry cask inner cavity PWR fuel shipments. Section D-5.1.2 provides the justification for use of the dose rates from the BWR fuel basket. From Section D-5.1.2, the analysis-basis assembly neutron source term is exceeded by a factor of 2.55

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ATTACHMENT 3

APPENDIX 9.3

D-9.3 Appendix C - Justification for Increase in Uranium Loading from 437 to 442 kg/assembly for the GIII PWR Fuel

This appendix addresses GIII PWR fuel assemblies with a maximum of 442 kg of uranium per assembly because it was discovered that some of the GIII PWR fuel had slightly greater uranium masses than the 437 kg originally analyzed. It addresses the impact of the slight increase in uranium mass on the structural, thermal, containment, shielding, and criticality assessments for the IF-300 Package. It is concluded that all of the transportation requirements specified in 10 CFR 71, as well as the limits specified in the IF-300 C of C (C of C No. 9001) and the Consolidated Safety Analysis Report (CSAR 1995) are satisfied when shipping GIII PWR fuel assemblies with uranium masses up to 442 kg/assembly as long as the other loading restrictions listed in Table D-1.1 are satisfied. Each of the assessments are addressed separately below.

D-9.3.1 Structural Assessment

An increase in uranium weight from 437 to 442 kg/assembly results in an overall weight of 658.2 kg for a GIII PWR fuel assembly, which is equal to a total fuel assembly loading of 3,949 kg (6 x 658.2 kg) for the PWR cask. This is below the structural evaluation design basis assembly loading of 4,920 kg for the PWR cask (see Section D-2.1). Therefore, there is no change in the structural aspects of the contents to be shipped.

D-9.3.2 Thermal Assessment

This section determines the decay heat for the GIII PWR fuel with a uranium loading of 442 kg/assembly. The calculation of the decay heat is performed using ORIGEN2 (Savino 1999). Section D-3.1 describes the ORIGEN2 models for the GIII PWR with a uranium loading of 437 kg/assembly for both the minimum initial enrichment (3.45 wt% U-235) and the maximum initial (4.25 wt% U-235) enrichment fuel. The only change made to the

ORIGEN2 models for this evaluation was to increase the uranium loading to 442 kg/assembly, which is approximately a 1.1% increase. These ORIGEN2 models were also used to determine the releasable source terms for the containment assessment in Appendix D-9.3.3, and the neutron and gamma source terms for the shielding assessment in Appendix D-9.3.4. The ORIGEN2 input files for both cases are included in Appendix D-9.4. ORIGEN2 was run on Duratek Federal Services Northwest Computer Serial Number 0022738297.

The increase in uranium results in an increase in the decay heat from 1,397 kW (4,768 Btu/h) to 1,414 kW (4,826 Btu/h) per assembly after a 5 year cooling time for the GIII PWR fuel assemblies with a minimum initial enrichment of 3.45 wt% ²³⁵U, which bounds that for the maximum initial enrichment of 4.25 wt% ²³⁵U. This corresponds to a total cask inner cavity heat load of:

$$\begin{aligned} \text{Total Cask Inner Cavity Heat Load} &= 6 \text{ FA/Cask} \times \\ &\quad 4,826 \\ &\quad \text{Btu/h/FA} \\ &= 28,956 \text{ Btu/h} \end{aligned}$$

It is concluded that the thermal analysis, which calculated temperatures and pressures in the IF-300 cask for a 40,000 Btu/h heat load, will bound the GIII PWR fuel with a minimum cooling time of 5 years and a maximum uranium loading of 442 kg/assembly.

D-9.3.3 Containment Assessment

This section addresses the impact of the change in the uranium mass loading for the GIII PWR fuel from 437 to 442 kg/assembly on the containment evaluation. The same ORIGEN2 models that were used to determine the decay heat in Section D-9.3.2 were used to calculate the releasable source terms under accident conditions for the containment assessment. NUREG/CR-6487 (NRC 1996) identifies 3 sources which should be considered

when determining the releasable source term for packages designed to transport irradiated fuel rods: 1) crud spallation from fuel rods, 2) release of fines from cladding breaches, and 3) source activity from gases and volatiles released due to cladding breaches. The following sections demonstrate that the analysis basis PWR fuel assembly source terms for each of these sources bound the source terms for the GIII PWR fuel with a uranium loading of 442 kg/assembly.

D-9.3.3.1 Crud Spallation from Fuel Rods

Following the methodology in NUREG/CR-6487 (NRC 1996) the crud spallation fraction (f_c) is a function of the surface area of the fuel rod and is independent of the uranium mass loading. Because the increased uranium loading does not affect the surface area of the fuel rod, the conclusion reached in Section D-4.1.2.1 for the GIII PWR fuel remains valid, i.e., that the crud spallation source term for the GIII PWR fuel is bounded by that for the analysis basis PWR fuel.

D-9.3.3.2 Source Activity due to Releases of Fines from Cladding Breaches

Section D-4.1.2.2 states that the only fuel-dependent parameters in the equation for the activity concentration inside the containment vessel as a result of fines released from cladding breaches are the total cask fuel mass and the specific activity of the fines (Ci/g).

Table D-9.3-1 lists the ORIGEN2 radioisotope inventory from Table D-4.1-2-2 for a single analysis basis PWR assembly with a 2 year cooling time and the inventory for a single GIII PWR assembly after a 5 year cooling time using the ORIGEN2 model developed in Section D-9.3.1 for a uranium loading of 442 kg/assembly. Note that the 3.45 wt% fuel inventory bounds that for the 4.25 wt% fuel. These results indicate that the analysis basis PWR fuel has a specific activity of 1.35 Ci/g ($6.30E5$ Ci/465,000 g) for a 2 year cooling time, and the GIII PWR fuel has a

specific activity of 0.82 Ci/g ($3.64E5$ Ci/442,000 g) for a 5 year cooling time.

The total cask fuel mass for a cask loading of 7 analysis basis GI 15x15 PWR fuel assemblies is 3,255 kg (7 x 465 kg) (see Section D-4.1.2.2) which is much greater than that for a cask loading of 2,652 kg (6 x 442 kg) for 6 GIII PWR fuel assemblies. Because both the specific activity and total cask fuel mass for the GIII PWR fuel are much lower than that for the analysis basis PWR fuel, the activity of the fuel fines for the analysis basis PWR fuel bounds that for the GIII PWR fuel.

D-9.3.3.3 Source Activity from Gases and Volatiles Released due to Cladding Breaches

The radioisotope inventory of the gases and volatiles from ORIGEN2 is listed in Table D-9.3-2 for the analysis basis fuel and for the GIII PWR fuel with a uranium loading of 442 kg/assembly. The values for the analysis basis fuel after a 2 year cooling time were taken from Table D-4.1-2-3. This comparison indicates that the GIII PWR fuel cooled 5 years has a lower inventory of gases and volatiles than the analysis basis PWR fuel assembly cooled 2 years. Therefore, the activity of the gases and volatiles from the analysis basis PWR fuel bounds that for the GIII PWR fuel with a uranium loading of 442 kg/assembly. Note that the radioisotope inventory for the gases from the 4.25% GIII PWR fuel bounds that for the 3.45% fuel, and the radioisotope inventory for the volatiles from the 3.45% GIII PWR fuel bounds that for the 4.25% fuel.

Table D-9.3-1

Radioisotope Inventories of Analysis-Basis and Group III
PWR Fuel to Determine the Specific Activity of the Fuel
(2 sheets total)

Analysis Basis PWR - 3.5%, 2 yr cooled		Group III PWR - 3.45%, 5 yr cooled	
Isotope	Ci ^a	Isotope	Ci ^a
H 3	3.46E+02	H 3	3.77E+02
FE55	2.72E+00	MN54	2.48E+00
CO60	5.79E+01	FE55	2.55E+02
ZN65	7.61E+00	CO60	9.90E+02
KR85	5.34E+03 ^b	NI63	6.55E+01
SR89	1.93E+01	KR85	4.02E+03
SR90	3.49E+04	SR90	3.85E+04
Y 90	3.49E+04	Y 90	3.85E+04
Y 91	9.93E+01	ZR93	1.13E+00
ZR95	2.91E+02	TC99	8.14E+00
NB95	6.68E+02	RU106	1.15E+04
NB95M	2.16E+00	RH106	1.15E+04
RU103	2.02E+00	AG110M	2.74E+01
RH103M	1.83E+00	CD113M	3.97E+01
RU106	6.36E+04	SN119M	2.74E+01
RH106	6.36E+04	SB125	3.15E+03
AG110	3.33E+00	TE125M	7.68E+02
AG110M	2.50E+02	CS134	2.82E+04
SN123	3.41E+01	CS137	6.11E+04
SB125	4.01E+03	BA137M	5.78E+04
TE125M	9.78E+02	CE144	6.27E+03
TE127	6.70E+01	PR144	6.27E+03
TE127M	6.84E+01	PR144M	7.52E+01
CS134	4.07E+04	PM147	1.43E+04
CS137	4.87E+04	SM151	2.36E+02
BA137M	4.61E+04	EU152	2.73E+00
CE144	9.19E+04	EU154	5.87E+03
PR144	9.19E+04	EU155	2.97E+03
PR144M	1.10E+03	U 237	1.51E+00
PM147	3.78E+04	NP239	2.52E+01
SM151	1.77E+02	PU238	3.29E+03
EU154	3.85E+03	PU239	1.93E+02
EU155	2.17E+03	PU240	3.45E+02
U 237	1.28E+00	PU241	6.16E+04
NP239	8.50E+00	PU242	1.48E+00
PU238	1.31E+03	AM241	6.44E+02
PU239	1.55E+02	AM242M	1.18E+01
PU240	2.35E+02	AM242	1.17E+01
PU241	5.23E+04	AM243	2.52E+01
AM241	2.33E+02	CM242	2.58E+01
AM242M	6.83E+00	CM243	2.48E+01

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Analysis Basis PWR - 3.5%, 2 yr cooled		Group III PWR - 3.45%, 5 yr cooled	
Isotope	Ci ^a	Isotope	Ci ^a
AM242	6.80E+00	CM244	4.86E+03
AM243	8.50E+00	Total	3.64E+05
CM242	8.82E+02	Specific Activity, Ci/g	0.82 ^c
CM243	8.60E+00		
CM244	9.54E+02		
Total	6.30E+05		
Specific Activity, Ci/g	1.35 ^c		

^a Note that isotopes with activities less than 1 Ci are not included since they have a negligible impact on the total.

^b The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 5.34E+03 Ci (13,086 Ci/0.35/7) for the analysis basis loading of 7 PWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

^c Specific activity is based on a U mass loading of 465 kg for the analysis basis PWR fuel and 442 kg for the SPC PWR fuel.

Table D-9.3-2

Comparison of Analysis-Basis PWR Fuel and
Group III PWR Fuel Gases and Volatiles

Isotope	Gases, Ci		Isotope	Volatiles, Ci	
	AB	Group III ^a		AB	Group III ^b
H 3	3.46E+02	3.63E+02	SR89	1.93E+01	4.42E-06
KR85	5.34E+03 ^c	4.29E+03	SR90	3.49E+04	3.85E+04
XE133	1.47E-36	0.00E+00	RU103	2.02E+00	8.64E-09
Total	5.69E+03	4.65E+03	RU106	6.36E+04	1.15E+04
			CS134	4.07E+04	2.82E+04
			CS136	4.49E-13	4.19E-38
			CS137	4.87E+04	6.11E+04
			Total	1.88E+05	1.39E+05

^a The 4.25% GIII PWR fuel has the largest inventory of gases.

^b The 3.45% GIII PWR fuel has the largest inventory of volatiles.

^c The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 5.34E+03 Ci (13,086 Ci/0.35/7) for the analysis basis loading of 7 PWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

D-9.3.4 Shielding Assessment

The same ORIGEN2 models that were used to determine the decay heat in Section D-9.3.2 were used to calculate the neutron and gamma source terms for the shielding assessment. Bounding neutron and gamma source terms occur for the minimum initial enrichment of 3.45 wt% U-235.

D-9.3.4.1 Neutron Source Strength Calculation

The total neutron emission rate from ORIGEN2 for GIII PWR fuel with a uranium loading of 442 kg/assembly is $6.88E+08$ neutrons/sec/assembly for 3.45 wt% U-235 at a burnup of 50 GWD/MTU and a decay time of 5 years. Therefore, the total neutron source emission rate for the cask loaded with 6 GIII PWR fuel assemblies is:

$$\begin{aligned}\text{Total Neutron Emission Rate} &= 6.88E+08 \text{ n/s/FA} \times \\ &\quad 6 \text{ FA/cask} \\ &= 4.13E+09 \text{ n/s/cask}\end{aligned}$$

D-9.3.4.2 Gamma Source Strength Calculation

The total gamma emission rate from ORIGEN2 for the bounding case is $9.35E+15$ gamma/sec/assembly. Therefore, the total gamma emission rate for the cask loaded with 6 GIII PWR fuel assemblies is:

$$\begin{aligned}\text{Total Gamma Emission Rate} &= 9.35E+15 \text{ } \gamma/\text{s/FA} \times 6 \\ &\quad \text{FA/cask} \\ &= 5.61E+16 \text{ } \gamma/\text{s/cask}\end{aligned}$$

D-9.3.4.3 Resulting Total Dose Rates

Tables D-9.3-3 and D-9.3-6 summarize the resulting dose rates for the increased uranium loading along with those for the current licensed design as listed in Tables D-7.1-2 through D-7.1-5. The dose rates shown for the GIII PWR Fuel with a uranium loading of 442 kg/assembly are calculated using the methodology described in Section D-7.1. Although the neutron dose rates

are higher than the analysis-basis values, the total neutron plus gamma dose rates are below the 10 CFR 71 limits and satisfy the limits stated in the existing C of C.

Table D-9.3-3

IF-300 PWR Cask NCT Dose Rates on the Cask Surface

Location	CURRENT LICENSED DESIGN (CSAR 1995), mrem/h			SIX GROUP III PWR ASSEMBLIES WITH 442 KG URANIUM LOADING, mrem/h		
	Neutron	Gamma	Total n+g	Neutron ¹	Gamma ²	Total n+g
Side(cask midplane), R _o	3.03	9.77	12.8	7.82	8.11	15.93
Side(top nozzle), R' _o	44.93	674.6	719.53	115.92	559.92	675.84
Top, T _o	44.93	13.7	58.63	115.92	11.37	127.29
Bottom, B _o	74.11	3.96	78.07	191.20	3.29	194.49
Dose Rate Limit			1000			1000

¹ Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly neutron source term to the analysis basis assembly neutron source term (from Section D-9.3.4.1, $2.58 = 6.88E8/2.67E8$ n/s).

² Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the assembly gamma source term for GIII PWR assemblies to that for the analysis basis fuel (from Section D-9.3.4.2, $0.83 = 9.35E15/1.13E16$ g/s).

Table D-9.3-4

IF-300 PWR Cask Estimated NCT Dose Rates at the Aluminum Enclosure and the Edge of the Conveyance Bed

Location	mrem/h
Side(cask midplane)	7.2 ¹
Side(top nozzle)	148 ¹
Top	<127.3 ²
Bottom	<194.5 ²
Dose Rate Limit	200

¹ Estimated dose rate at the top of the aluminum enclosure adjacent to the cask midplane or top nozzle. See Section D-7.1-1 for the calculational methodology.

² Estimated dose rate at the edge of the conveyance bed for the top and bottom of the cask. See Section D-7.1-1 for the calculational methodology.

Table D-9.3-5

IF-300 PWR Cask NCT Dose Rates at 2 m from the Vehicle Surface

Location	CURRENT LICENSED DESIGN (CSAR 1995), mrem/h			SIX GROUP III PWR ASSEMBLIES WITH 442 KG URANIUM LOADING mrem/h		
	Neutron	Gamma	Total n+g	Neutron ¹	Gamma ²	Total n+g
Side(cask midplane), R _o	0.71	2.33	3.04	1.83	1.93	3.77
Side(top nozzle), R' _o	0.92	8.5	9.42	2.37	7.06	9.43
Top, T _o	1.8	2.82	4.62	4.64	2.34	6.98
Bottom, B _o	2.97	0.95	3.92	7.66	0.79	8.45
Dose Rate Limit			10			10

¹ Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly neutron source term to the analysis basis assembly neutron source term (from Section D-9.3.4.1, $2.58 = 6.88E8/2.67E8$ n/s)

² Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the assembly gamma source term for GIII assemblies to that for the analysis basis fuel (from Section D-9.3.4.2, $0.83 = 9.35E15/1.13E16$ n/s).

Table D-9.3-6

IF-300 PWR Cask Accident Condition Dose Rates at 1 m from the Cask

LOCATION	CURRENT LICENSED DESIGN (CSAR 1995), mrem/h			SIX GROUP III PWR ASSEMBLIES WITH 442 KG URANIUM LOADING mrem/h		
	NEUTRON	GAMMA	TOTAL	NEUTRON ¹	GAMMA ²	TOTAL
RADIAL CL	163.06	18.10	181.16	420.69	15.02	435.72
10 CFR 71 LIMIT			1000			1000

¹ Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly neutron source term to the analysis basis assembly neutron source term (from Section D-9.3.4.1, $2.58 = 6.88E8/2.67E8$ n/s).

² Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the assembly gamma source term for GIII assemblies to that for the analysis basis fuel (from Section D-9.3.4.2, $0.83 = 9.35E15/1.13E16$ g/s).

D-9.3.5 Criticality Assessment

In order to demonstrate that the small increase in uranium loading from 437 to 442 kg/assembly has a negligible impact on the criticality assessment for the IF-300 Cask, two MCNP criticality cases from Section 6 were rerun with the increased uranium loading. The MCNP input files for both cases are included in Appendix D-9.4. MCNP was run on Duratek Federal Services Northwest Computer Serial Number 0022738297.

The first case is for a single IF-300 containing six GIII PWR fuel assemblies with a maximum lattice average enrichment of 4.25 wt% ²³⁵U loaded in the peripheral PWR basket locations (i.e., with the center basket location empty) and with a 12 in close-fitting water reflector around the cask. This case is identified as case Pso6-10 in Table D-6.4-3 and resulted in the highest k_{tot} (0.94427) for the single cask cases. The MCNP input file for case Pso6-10 was modified by slightly increasing the UO₂ density in the fuel to yield a 442 kg uranium loading per assembly. The remaining input for the MCNP model was unchanged. This case (Pso6-10a) resulted in a k_{tot} of 0.94317, which is within 1 standard deviation ($\sigma = 0.00117$) of the k_{tot} for the original case (0.94427).

In Section D-6.6.1, a series of array cases were run with the maximum k_{tot} value of 0.94636 occurring for MCNP case pso6i10a (see Table D-6.6-1). This case is for an infinite array of IF-300 casks containing six GIII PWR fuel assemblies with a maximum lattice average enrichment of 4.25 wt% ²³⁵U loaded in the peripheral PWR basket locations (i.e., with the center basket location empty). The MCNP input file for case pso6i10a was modified by slightly increasing the UO₂ density in the fuel to produce a 442 kg uranium loading per assembly. The remaining input for the MCNP model was unchanged. This case (pso6i10z) resulted in a k_{tot} of 0.94489, which is within 2 standard deviations ($\sigma = 0.00104$) of the k_{tot} for the original case (0.94636).

It is concluded that the increase in uranium mass loading from 437 to 442 kg/assembly has a negligible impact on the criticality assessment originally performed in Section D-6 because the differences between the k_{tot} values are statistically insignificant. Therefore, both a single IF-300 Cask and an infinite array of IF-300 Casks containing six 4.25 wt% ^{235}U GIII PWR fuel assemblies with a uranium loading up to 442 kg/assembly and with the center basket location empty will meet the criticality safety limit of $k_{eff} < 0.95$, which results in a criticality TI=0.

D-9.4 ORIGEN2 Input files for 442 kg/assembly
for the GIII PWR Fuel

```

-1
-1
-1
RDA #####
RDA ## IF300 HIGH BURNUP/ENRICHMENT PWR FUEL ##
RDA ## Input Filename: PWR345UE ORIGEN2 VER2.1 ##
RDA ## Creation Date: 7/23/02 ##
RDA ## Fuel Type: 3.45% W 15X15 ANF PWR ##
RDA ## Burnup:50000 Mwd/MTIHM, SP.POWER = 40MW/MTU ##
RDA ## Cycle Burnup: 17.5, 17.5, 15K, PWRUE xsect lib ##
RDA ## CPL-WEIGHT,IRR HISTORY,FLX FACT=DOE/RW 0184,RV1 ##
RDA ## Low End Enrichment for Shielding Source Term ##
RDA #####
LIP 0 0 0
LIB 0 1 2 3 604 605 606 9 0 0 1 39
PHO 101 102 103 10
RDA READ UNIT AMOUNTS OF FUEL AND FUEL ASSEMBLY MATERIALS
RDA -1 = FRESH PWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA -2 = FRESH ZIRCALOY COMPOSITION (1 KG)
RDA -3 = FRESH SS 304 COMPOSITION (1 KG)
RDA -4 = FRESH INCONEL-X750 COMPOSITION (1 KG)
RDA -5 = FRESH INCONEL-718 COMPOSITION (1 KG)
TIT INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP -1 1 -1 -1 1 1
RDA READ ZIRCALOY COMPOSITION (1.0 KG)
INP -2 1 -1 -1 1 1
RDA READ SS304 COMPOSITION (1.0 KG)
INP -3 1 -1 -1 1 1
RDA READ INCONEL X750 COMPOSITION (1.0 KG)
INP -4 1 -1 -1 1 1
RDA READ INCONEL 718 COMPOSITION (1.0 KG)
INP -5 1 -1 -1 1 1
RDA MIX TOP, PLENUM, IN-CORE, AND BOTTOM MIXTURES
RDA MIX TOP ZONE
MOV -3 1 0 6.381 SS304
ADD -4 1 0 0.024 INCONEL-X750
ADD -5 1 0 0.672 INCONEL-718
RDA MIX PLENUM ZONE
MOV -4 2 0 3.019 INCONEL-X750
ADD -2 2 0 7.551 ZIRCALOY-4
RDA MIX IN-CORE ZONE WITHOUT UO2
MOV -5 3 0 0.112 INCONEL-718
ADD -2 3 0 133.927 ZIRCALOY-4
RDA MIX BOTTOM ZONE
MOV -3 4 0 4.459 SS304
RDA MOVE MIXTURES INTO SCRATCH VECTORS
MOV -1 -5 0 1.0 IN CORE U ONLY
MOV 1 -1 0 1.0 TOP ZONE
MOV 2 -2 0 1.0 PLENUM ZONE
MOV 3 -3 0 1.0 INCORE W/O UO2
MOV 4 -4 0 1.0 BOTTOM NOZZLE
RDA IRRADIATION OF ONE METRIC TON OF PWRU FUEL AT 100% POWER
BUP
IRP 75 40.0 -5 1 4 2 BURNUP= 3,000 MWD/MTIHM
IRP 150 40.0 1 2 4 0 BURNUP= 6,000 MWD/MTIHM
IRP 225 40.0 2 3 4 0 BURNUP= 9,000 MWD/MTIHM
IRP 300 40.0 3 4 4 0 BURNUP=12,000 MWD/MTIHM
IRP 375 40.0 4 5 4 0 BURNUP=15,000 MWD/MTIHM
IRP 437.5 40.0 5 5 4 0 BURNUP=17,500 MWD/MTIHM
DEC 543.5 5 6 4 0 DECAY FOR 106.0 DAYS
IRP 618.5 40.0 6 7 4 0 BURNUP=20,500 MWD/MTIHM
IRP 693.5 40.0 7 8 4 0 BURNUP=23,500 MWD/MTIHM
IRP 768.5 40.0 8 9 4 0 BURNUP=26,500 MWD/MTIHM

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IRP	843.5	40.0	9	10	4	0	BURNUP=29,500 MWD/MTIHM
IRP	918.5	40.0	10	11	4	0	BURNUP=32,500 MWD/MTIHM
IRP	981	40.0	11	11	4	0	BURNUP=35,000 MWD/MTIHM
DEC	1087		11	12	4	0	DECAY FOR 106.0 DAYS
IRP	1162	40.0	12	1	4	0	BURNUP=38,000 MWD/MTIHM
IRP	1237	40.0	1	2	4	0	BURNUP=41,000 MWD/MTIHM
IRP	1312	40.0	2	3	4	0	BURNUP=44,000 MWD/MTIHM
IRP	1387	40.0	3	4	4	0	BURNUP=47,000 MWD/MTIHM
IRP	1462	40.0	4	5	4	0	BURNUP=50,000 MWD/MTIHM
BUP							
RDA	-5	= IRRADIATED UO2 FUEL AT DISCHARGE 0.442 MTU/FA					
MOV	5	-5	0	0.442			
RDA							
RDA		IRRADIATION OF INCORE MATERIAL WITHOUT UO2 AT 100% FLUX					
IRF	75.0	-1.0	-3	1	4	2	BURNUP= 3,000 MWD/MTIHM
IRF	150	-1.0	1	2	4	0	BURNUP= 6,000 MWD/MTIHM
IRF	225	-1.0	2	3	4	0	BURNUP= 9,000 MWD/MTIHM
IRF	300	-1.0	3	4	4	0	BURNUP=12,000 MWD/MTIHM
IRF	375	-1.0	4	5	4	0	BURNUP=15,000 MWD/MTIHM
IRF	437.5	-1.0	5	5	4	0	BURNUP=17,500 MWD/MTIHM
DEC	543.5		5	6	4	0	DECAY FOR 106.0 DAYS
IRF	618.5	-1.0	6	7	4	0	BURNUP=20,500 MWD/MTIHM
IRF	693.5	-1.0	7	8	4	0	BURNUP=23,500 MWD/MTIHM
IRF	768.5	-1.0	8	9	4	0	BURNUP=26,500 MWD/MTIHM
IRF	843.5	-1.0	9	10	4	0	BURNUP=29,500 MWD/MTIHM
IRF	918.5	-1.0	10	11	4	0	BURNUP=32,500 MWD/MTIHM
IRF	981	-1.0	11	11	4	0	BURNUP=35,000 MWD/MTIHM
DEC	1087		11	12	4	0	DECAY FOR 106.0 DAYS
IRF	1162	-1.0	12	1	4	0	BURNUP=38,000 MWD/MTIHM
IRF	1237	-1.0	1	2	4	0	BURNUP=41,000 MWD/MTIHM
IRF	1312	-1.0	2	3	4	0	BURNUP=44,000 MWD/MTIHM
IRF	1387	-1.0	3	4	4	0	BURNUP=47,000 MWD/MTIHM
IRF	1462	-1.0	4	-3	4	0	BURNUP=50,000 MWD/MTIHM
RDA							
RDA	-3	= IRRADIATED INCORE MATERIAL WITHOUT UO2					
RDA							
RDA		IRRADIATION OF TOP NOZZLE MATERIAL AT 10% FLUX					
RDA							
IRF	75.0	-0.1	-1	1	4	2	BURNUP= 3,000 MWD/MTIHM
IRF	150	-0.1	1	2	4	0	BURNUP= 6,000 MWD/MTIHM
IRF	225	-0.1	2	3	4	0	BURNUP= 9,000 MWD/MTIHM
IRF	300	-0.1	3	4	4	0	BURNUP=12,000 MWD/MTIHM
IRF	375	-0.1	4	5	4	0	BURNUP=15,000 MWD/MTIHM
IRF	437.5	-0.1	5	5	4	0	BURNUP=17,500 MWD/MTIHM
DEC	543.5		5	6	4	0	DECAY FOR 106.0 DAYS
IRF	618.5	-0.1	6	7	4	0	BURNUP=20,500 MWD/MTIHM
IRF	693.5	-0.1	7	8	4	0	BURNUP=23,500 MWD/MTIHM
IRF	768.5	-0.1	8	9	4	0	BURNUP=26,500 MWD/MTIHM
IRF	843.5	-0.1	9	10	4	0	BURNUP=29,500 MWD/MTIHM
IRF	918.5	-0.1	10	11	4	0	BURNUP=32,500 MWD/MTIHM
IRF	981	-0.1	11	11	4	0	BURNUP=35,000 MWD/MTIHM
DEC	1087		11	12	4	0	DECAY FOR 106.0 DAYS
IRF	1162	-0.1	12	1	4	0	BURNUP=38,000 MWD/MTIHM
IRF	1237	-0.1	1	2	4	0	BURNUP=41,000 MWD/MTIHM
IRF	1312	-0.1	2	3	4	0	BURNUP=44,000 MWD/MTIHM
IRF	1387	-0.1	3	4	4	0	BURNUP=47,000 MWD/MTIHM
IRF	1462	-0.1	4	-1	4	0	BURNUP=50,000 MWD/MTIHM
RDA							
RDA	-1	= IRRADIATED TOP NOZZLE MATERIAL					
RDA							
RDA		IRRADIATION OF PLENUM ZONE MATERIAL AT 20% FLUX					
RDA							
IRF	75.0	-0.2	-2	1	4	2	BURNUP= 3,000 MWD/MTIHM
IRF	150	-0.2	1	2	4	0	BURNUP= 6,000 MWD/MTIHM
IRF	225	-0.2	2	3	4	0	BURNUP= 9,000 MWD/MTIHM
IRF	300	-0.2	3	4	4	0	BURNUP=12,000 MWD/MTIHM
IRF	375	-0.2	4	5	4	0	BURNUP=15,000 MWD/MTIHM
IRF	437.5	-0.2	5	5	4	0	BURNUP=17,500 MWD/MTIHM
DEC	543.5		5	6	4	0	DECAY FOR 106.0 DAYS

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```

IRF 618.5 -0.2 6 7 4 0 BURNUP=20,500 MWD/MTIHM
IRF 693.5 -0.2 7 8 4 0 BURNUP=23,500 MWD/MTIHM
IRF 768.5 -0.2 8 9 4 0 BURNUP=26,500 MWD/MTIHM
IRF 843.5 -0.2 9 10 4 0 BURNUP=29,500 MWD/MTIHM
IRF 918.5 -0.2 10 11 4 0 BURNUP=32,500 MWD/MTIHM
IRF 981 -0.2 11 11 4 0 BURNUP=35,000 MWD/MTIHM
DEC 1087 11 12 4 0 DECAY FOR 106.0 DAYS
IRF 1162 -0.2 12 1 4 0 BURNUP=38,000 MWD/MTIHM
IRF 1237 -0.2 1 2 4 0 BURNUP=41,000 MWD/MTIHM
IRF 1312 -0.2 2 3 4 0 BURNUP=44,000 MWD/MTIHM
IRF 1387 -0.2 3 4 4 0 BURNUP=47,000 MWD/MTIHM
IRF 1462 -0.2 4 -2 4 0 BURNUP=50,000 MWD/MTIHM

```

RDA

-2 = IRRADIATED PLENUM ZONE MATERIAL

RDA

IRRADIATION OF BOTTOM NOZZLE MATERIAL AT 20% FLUX

RDA

```

IRF 75.0 -0.2 -4 1 4 2 BURNUP= 3,000 MWD/MTIHM
IRF 150 -0.2 1 2 4 0 BURNUP= 6,000 MWD/MTIHM
IRF 225 -0.2 2 3 4 0 BURNUP= 9,000 MWD/MTIHM
IRF 300 -0.2 3 4 4 0 BURNUP=12,000 MWD/MTIHM
IRF 375 -0.2 4 5 4 0 BURNUP=15,000 MWD/MTIHM
IRF 437.5 -0.2 5 5 4 0 BURNUP=17,500 MWD/MTIHM
DEC 543.5 5 6 4 0 DECAY FOR 106.0 DAYS
IRF 618.5 -0.2 6 7 4 0 BURNUP=20,500 MWD/MTIHM
IRF 693.5 -0.2 7 8 4 0 BURNUP=23,500 MWD/MTIHM
IRF 768.5 -0.2 8 9 4 0 BURNUP=26,500 MWD/MTIHM
IRF 843.5 -0.2 9 10 4 0 BURNUP=29,500 MWD/MTIHM
IRF 918.5 -0.2 10 11 4 0 BURNUP=32,500 MWD/MTIHM
IRF 981 -0.2 11 11 4 0 BURNUP=35,000 MWD/MTIHM
DEC 1087 11 12 4 0 DECAY FOR 106.0 DAYS
IRF 1162 -0.2 12 1 4 0 BURNUP=38,000 MWD/MTIHM
IRF 1237 -0.2 1 2 4 0 BURNUP=41,000 MWD/MTIHM
IRF 1312 -0.2 2 3 4 0 BURNUP=44,000 MWD/MTIHM
IRF 1387 -0.2 3 4 4 0 BURNUP=47,000 MWD/MTIHM
IRF 1462 -0.2 4 -4 4 0 BURNUP=50,000 MWD/MTIHM

```

RDA

-4 = IRRADIATED BOTTOM NOZZLE MATERIAL

RDA

ADD VECTORS 1-5 TO CREATE A WHOLE FUEL ASSEMBLY

```

MOV -1 -6 0 1.0
ADD -2 -6 0 1.0
ADD -3 -6 0 1.0
ADD -4 -6 0 1.0
ADD -5 -6 0 1.0
MOV -6 -7 0 1.0

```

RDA

1 = IRRADIATED WHOLE ASSEMBLY AT DISCHARGE

TIT SOURCE FOR 3.45%, 50 GWD/MTU PWR FUEL AT 0.442 MTU/FA

```

DEC 1 -7 1 1 2
DEC 1 1 2 5 0
DEC 2 2 3 5 0
DEC 3 3 4 5 0
DEC 4 4 5 5 0
DEC 5 5 6 5 0

```

BAS ONE W 15X15 ANF PWR FUEL ASSEMBLY

```

OPTL 8 8 8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTA 8 8 8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF 8 8 8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
CUT 7 1.0e-8 9 0.01 25 0.01 25 0.01 27 0.01 -1
OUT 6 1 -1 0
STP 4

```

```

2 922340 89.0 922350 34500.0 922360 46.0 922380 965365.0 FUEL 3.45%
4 030000 1.0 050000 1.0 060000 89.4 070000 25.0 FUEL IMPU
4 080000 134454. 090000 10.7 110000 15.0 120000 2.0 FUEL IMPU
4 130000 16.7 140000 12.1 150000 35.0 170000 5.3 FUEL IMPU
4 200000 2.0 220000 1.0 230000 3.0 240000 4.0 FUEL IMPU
4 250000 1.7 260000 18.0 270000 1.0 280000 24.0 FUEL IMPU
4 290000 1.0 300000 40.3 420000 10.0 470000 0.1 FUEL IMPU
4 480000 25.0 490000 2.0 500000 4.0 640000 2.5 FUEL IMPU

```

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```

4 740000 2.0 820000 1.0 830000 0.4 0 0.0 FUEL IMPU
0
4 400000 979.069 500000 16.0 260000 2.25 240000 1.25 ZIRC-4
4 280000 0.02 130000 0.024 050000 0.00033 480000 0.00025 ZIRC-4
4 060000 0.120 270000 0.010 290000 0.020 720000 0.078 ZIRC-4
4 010000 0.013 250000 0.020 070000 0.080 080000 0.950 ZIRC-4
4 160000 0.035 220000 0.020 740000 0.020 230000 0.020 ZIRC-4
5 920000 0.0002 410000 0.12 0 0.0 0 0.0 ZIRC-4
0
4 260000 687.15 240000 190.0 280000 89.2 250000 20.0 SS-304
4 060000 0.8 150000 0.45 160000 0.3 140000 10.0 SS-304
4 070000 1.3 270000 0.8 410000 0.12 0 0.0 SS-304
0
4 260000 67.846 240000 149.660 280000 721.859 130000 7.982 INC-750
4 060000 0.399 270000 6.485 290000 0.499 250000 6.984 INC-750
4 420000 0.0000 070000 1.3 410000 8.980 160000 0.07 INC-750
4 140000 2.993 220000 24.943 0 0.0 0 0.0 INC-750
0
4 260000 179.766 240000 189.753 280000 519.623 130000 5.992 INC-718
4 060000 0.4 270000 4.694 290000 0.999 250000 1.997 INC-718
4 420000 29.961 070000 1.3 410000 55.458 160000 0.07 INC-718
4 140000 1.997 220000 7.99 0 0.0 0 0.0 INC-718
0
END

```

-1
-1
-1

```

RDA #####
RDA ## IF300 HIGH BURNUP/ENRICHMENT PWR FUEL ##
RDA ## Input Filename: P425UE ORIGEN2 VER2.1 ##
RDA ## Creation Date: 7/23/02 ##
RDA ## Fuel Type: 4.25% W 15X15 ANF PWR ##
RDA ## Burnup:50000 Mwd/MTIHM, SP.POWER = 40MW/MTU ##
RDA ## Cycle Burnup: 17.5, 17.5, 15K, PWRUE xsect lib ##
RDA ## CPL-WEIGHT,IRR HISTORY,FLX FACT=DOE/RW 0184,RV1 ##
RDA #####
LIP 0 0 0
LIB 0 1 2 3 604 605 606 9 0 0 1 39
PHO 101 102 103 10
RDA READ UNIT AMOUNTS OF FUEL AND FUEL ASSEMBLY MATERIALS
RDA -1 = FRESH PWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
TIT INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP -1 1 -1 -1 1 1
RDA IRRADIATION OF ONE METRIC TON OF PWRU FUEL AT 100% POWER
BUP
IRP 75 40.0 -1 1 4 2 BURNUP= 3,000 MWD/MTIHM
IRP 150 40.0 1 2 4 0 BURNUP= 6,000 MWD/MTIHM
IRP 225 40.0 2 3 4 0 BURNUP= 9,000 MWD/MTIHM
IRP 300 40.0 3 4 4 0 BURNUP=12,000 MWD/MTIHM
IRP 375 40.0 4 5 4 0 BURNUP=15,000 MWD/MTIHM
IRP 437.5 40.0 5 5 4 0 BURNUP=17,500 MWD/MTIHM
DEC 543.5 5 6 4 0 DECAY FOR 106.0 DAYS
IRP 618.5 40.0 6 7 4 0 BURNUP=20,500 MWD/MTIHM
IRP 693.5 40.0 7 8 4 0 BURNUP=23,500 MWD/MTIHM
IRP 768.5 40.0 8 9 4 0 BURNUP=26,500 MWD/MTIHM
IRP 843.5 40.0 9 10 4 0 BURNUP=29,500 MWD/MTIHM
IRP 918.5 40.0 10 11 4 0 BURNUP=32,500 MWD/MTIHM
IRP 981 40.0 11 11 4 0 BURNUP=35,000 MWD/MTIHM
DEC 1087 11 12 4 0 DECAY FOR 106.0 DAYS
IRP 1162 40.0 12 1 4 0 BURNUP=38,000 MWD/MTIHM
IRP 1237 40.0 1 -2 4 0 BURNUP=41,000 MWD/MTIHM
IRP 1312 40.0 -2 -3 4 0 BURNUP=44,000 MWD/MTIHM
IRP 1387 40.0 -3 -4 4 0 BURNUP=47,000 MWD/MTIHM
IRP 1462 40.0 -4 -5 4 0 BURNUP=50,000 MWD/MTIHM
BUP

```

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```

RDA      IRRADIATED UO2 FUEL - 0.442 MTU/FA
MOV      -5  -7    0    0.442
RDA
RDA      1 = IRRADIATED WHOLE ASSEMBLY AT DISCHARGE
TIT      SOURCE FOR 4.25%, 50 GWD/MTU PWR FUEL AT 0.442 MTU/FA
DEC      1      -7      1      1      2
DEC      1      1      2      5      0
DEC      2      2      3      5      0
DEC      3      3      4      5      0
DEC      4      4      5      5      0
DEC      5      5      6      5      0
BAS      ONE W 15X15 ANF PWR FUEL ASSEMBLY
HED      1 DISCHARGE
OPTL     8 8 8 8 7 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8
OPTA     8 8 8 8 7 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8
OPTF     8 8 8 8 7 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8
CUT      7 0.0001 9 0.01 25 0.01 25 0.01 27 0.01 -1
OUT      6 1 -1 0
STP      4
2  922340 89. 922350 42500. 922360 46. 922380 957365. FUEL 4.25%
4  030000 1.0 050000 1.0 060000 89.4 070000 25.0 FUEL IMPU
4  080000 134454. 090000 10.7 110000 15.0 120000 2.0 FUEL IMPU
4  130000 16.7 140000 12.1 150000 35.0 170000 5.3 FUEL IMPU
4  200000 2.0 220000 1.0 230000 3.0 240000 4.0 FUEL IMPU
4  250000 1.7 260000 18.0 270000 1.0 280000 24.0 FUEL IMPU
4  290000 1.0 300000 40.3 420000 10.0 470000 0.1 FUEL IMPU
4  480000 25.0 490000 2.0 500000 4.0 640000 2.5 FUEL IMPU
4  740000 2.0 820000 1.0 830000 0.4 0 0.0 FUEL IMPU
0
      END

```

D-9.5 MCNP Input files for 442 kg/assembly for the
GIII PWR Fuel

Case Pso6-10a

```
PWR Cask - 6 FAs - 4.25% - 1.00 g/cc - 12 in Refl - Shift out
1 1 -10.406 -1 22 -23 u=1 imp:n=1 $ Enriched uo2 fuel
2 4 -1.000 1 -2 22 -23 u=1 imp:n=1 $ gap - H2O
3 3 -6.565 2 -3 22 -23 u=1 imp:n=1 $ Zr4 Clad
4 4 -1.00 3 16 -17 18 -19 22 -23 u=1 imp:n=1 $ Moderator
501 4 -1.0 (-16:17:-18:19) 22 -467 u=1 imp:n=1 $ Spacer - Moderator
502 3 -6.565 (-16:17:-18:19) 467 -466 u=1 imp:n=1 $ Zr4 - 2nd Spacer
503 4 -1.0 (-16:17:-18:19) 466 -465 u=1 imp:n=1 $ Spacer - Moderator
504 3 -6.565 (-16:17:-18:19) 465 -464 u=1 imp:n=1 $ Zr4 - 3rd Spacer
505 4 -1.0 (-16:17:-18:19) 464 -463 u=1 imp:n=1 $ Spacer - Moderator
506 3 -6.565 (-16:17:-18:19) 463 -462 u=1 imp:n=1 $ Zr4 - 4th Spacer
507 4 -1.0 (-16:17:-18:19) 462 -461 u=1 imp:n=1 $ Spacer - Moderator
508 3 -6.565 (-16:17:-18:19) 461 -460 u=1 imp:n=1 $ Zr4 - 5th Spacer
509 4 -1.0 (-16:17:-18:19) 460 -459 u=1 imp:n=1 $ Spacer - Moderator
510 3 -6.565 (-16:17:-18:19) 459 -458 u=1 imp:n=1 $ Zr4 - 6th Spacer
511 4 -1.0 (-16:17:-18:19) 458 -457 u=1 imp:n=1 $ Spacer - Moderator
512 3 -6.565 (-16:17:-18:19) 457 -456 u=1 imp:n=1 $ Zr4 - 7th Spacer
513 4 -1.0 (-16:17:-18:19) 456 -455 u=1 imp:n=1 $ Spacer - Moderator
514 3 -6.565 (-16:17:-18:19) 455 -454 u=1 imp:n=1 $ Zr4 - 8th Spacer
515 4 -1.0 (-16:17:-18:19) 454 -453 u=1 imp:n=1 $ Spacer - Moderator
516 3 -6.565 (-16:17:-18:19) 453 -452 u=1 imp:n=1 $ Zr4 - 9th Spacer
517 4 -1.0 (-16:17:-18:19) 452 -23 u=1 imp:n=1 $ Spacer - Moderator
c Natural Uranium blanket - Top
5 5 -10.406 -1 23 -21 u=1 imp:n=1 $ Natural uo2 fuel
6 4 -1.000 1 -2 23 -21 u=1 imp:n=1 $ gap - H2O
7 3 -6.565 2 -3 23 -21 u=1 imp:n=1 $ Zr4 Clad
8 4 -1.0 3 23 -21 u=1 imp:n=1 $ Moderator
c Natural Uranium blanket - Bottom
9 5 -10.406 -1 20 -22 u=1 imp:n=1 $ Natural uo2 fuel
10 4 -1.000 1 -2 20 -22 u=1 imp:n=1 $ gap - H2O
11 3 -6.565 2 -3 20 -22 u=1 imp:n=1 $ Zr4 Clad
12 4 -1.0 3 20 -22 u=1 imp:n=1 $ Moderator
c Plenum region - below fuel pins
13 4 -1.000 -2 24 -20 u=1 imp:n=1 $ He
14 3 -6.565 2 -3 24 -20 u=1 imp:n=1 $ Zr4 Clad
15 4 -1.00 3 24 -20 u=1 imp:n=1 $ Moderator
c Plenum region - above fuel pins
16 4 -1.000 -2 21 -25 u=1 imp:n=1 $ He
17 3 -6.565 2 -3 21 -25 u=1 imp:n=1 $ Zr4 Clad
18 4 -1.00 3 21 -25 u=1 imp:n=1 $ Moderator
c Rest of Lower tie plate - below fuel pins
19 6 -8.03 30 -24 u=1 imp:n=1 $ 304L SS in LTP
c Water below Lower tie plate/above channel bottom -below fuel pins
191 4 -1.00 -30 u=1 imp:n=1 $ Water
c Rest of Upper tie plate - above fuel pins
20 6 -8.03 25 u=1 imp:n=1 $ 304L SS in LTP
c Lattice of Cells - Top of water above UTP to water below LTP
25 0 4 -5 6 -7 lat=1 u=2 fill= -7:7 -7:7 0:0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ x=-7 to x=7;y=7
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=6
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1 1 $ " ";y=5
1 1 1 1 1 1 1 1 3 1 1 1 1 1 1 1 $ " ";y=4
1 1 1 1 3 1 1 1 1 1 3 1 1 1 1 1 $ " ";y=3
1 1 3 1 1 1 1 1 1 1 1 1 3 1 1 1 $ " ";y=2
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=1
1 1 1 3 1 1 1 1 3 1 1 1 3 1 1 1 $ " ";y=0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=-1
1 1 3 1 1 1 1 1 1 1 1 1 3 1 1 1 $ " ";y=-2
1 1 1 1 3 1 1 1 1 1 3 1 1 1 1 1 $ " ";y=-3
1 1 1 1 1 1 1 1 3 1 1 1 1 1 1 1 $ " ";y=-4
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1 1 $ " ";y=-5
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=-6
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 imp:n=1 $ " ";y=-7
```

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```

c Instrument/Control Rod Guide Tubes
26 4 -1.00 -10 24 -25 u=3 imp:n=1 $ water inside tube
27 3 -6.565 10 -11 24 -25 u=3 imp:n=1 $ Zr4 Clad
28 4 -1.0 11 16 -17 18 -19 24 -25 u=3 imp:n=1 $ Moderator
281 4 -1.0 (-16:17:-18:19) 24 -467 u=3 imp:n=1 $ Spacer - Moderator
282 3 -6.565 (-16:17:-18:19) 467 -466 u=3 imp:n=1 $ Zr4 - 2nd Spacer
283 4 -1.0 (-16:17:-18:19) 466 -465 u=3 imp:n=1 $ Spacer - Moderator
284 3 -6.565 (-16:17:-18:19) 465 -464 u=3 imp:n=1 $ Zr4 - 3rd Spacer
285 4 -1.0 (-16:17:-18:19) 464 -463 u=3 imp:n=1 $ Spacer - Moderator
286 3 -6.565 (-16:17:-18:19) 463 -462 u=3 imp:n=1 $ Zr4 - 4th Spacer
287 4 -1.0 (-16:17:-18:19) 462 -461 u=3 imp:n=1 $ Spacer - Moderator
288 3 -6.565 (-16:17:-18:19) 461 -460 u=3 imp:n=1 $ Zr4 - 5th Spacer
289 4 -1.0 (-16:17:-18:19) 460 -459 u=3 imp:n=1 $ Spacer - Moderator
290 3 -6.565 (-16:17:-18:19) 459 -458 u=3 imp:n=1 $ Zr4 - 6th Spacer
291 4 -1.0 (-16:17:-18:19) 458 -457 u=3 imp:n=1 $ Spacer - Moderator
292 3 -6.565 (-16:17:-18:19) 457 -456 u=3 imp:n=1 $ Zr4 - 7th Spacer
293 4 -1.0 (-16:17:-18:19) 456 -455 u=3 imp:n=1 $ Spacer - Moderator
294 3 -6.565 (-16:17:-18:19) 455 -454 u=3 imp:n=1 $ Zr4 - 8th Spacer
295 4 -1.0 (-16:17:-18:19) 454 -453 u=3 imp:n=1 $ Spacer - Moderator
296 3 -6.565 (-16:17:-18:19) 453 -452 u=3 imp:n=1 $ Zr4 - 9th Spacer
297 4 -1.0 (-16:17:-18:19) 452 -25 u=3 imp:n=1 $ Spacer - Moderator
c Lower Tie Plate
c Instrument/Control Rod Guide Tube Locations
29 4 -1.00 -10 -24 30 u=3 imp:n=1 $ water inside tube
30 3 -6.565 10 -11 -24 30 u=3 imp:n=1 $ Zr4 Clad
31 6 -8.03 11 -24 30 u=3 imp:n=1 $ 304L SS in LTP
c Water below Lower tie plate/above channel bottom -below fuel pins
c 311 4 -1.00 -30 u=3 imp:n=1 $ Water
c Upper Tie Plate
c Instrument/Control Rod Guide Tube Locations
33 4 -1.00 -10 25 u=3 imp:n=1 $ water inside tube
34 3 -6.565 10 -11 25 u=3 imp:n=1 $ Zr4 Clad
35 6 -8.03 11 25 u=3 imp:n=1 $ 304L SS in UTP
c Fuel Assembly
c
36 0 12 -13 14 -15 26 -31 u=42 fill=2 trcl=132 imp:n=1 $ FA #2
363 0 12 -13 14 -15 26 -31 u=43 fill=2 trcl=133 imp:n=1 $ FA #3
364 0 12 -13 14 -15 26 -31 u=44 fill=2 trcl=134 imp:n=1 $ FA #4
365 0 12 -13 14 -15 26 -31 u=45 fill=2 trcl=135 imp:n=1 $ FA #5
366 0 12 -13 14 -15 26 -31 u=46 fill=2 trcl=136 imp:n=1 $ FA #6
367 0 12 -13 14 -15 26 -31 u=47 fill=2 trcl=137 imp:n=1 $ FA #7
c
c Water Gap between fuel assembly and guide channel
c and above UTP.
c
40 4 -1.0 (41 -42 45 -46 26) u=4 imp:n=1 $ FA #1 - Center Hole
402 4 -1.0 (41 -42 45 -46 26) #36 u=42 imp:n=1 $ FA #2
403 4 -1.0 (41 -42 45 -46 26) #363 u=43 imp:n=1 $ FA #3
404 4 -1.0 (41 -42 45 -46 26) #364 u=44 imp:n=1 $ FA #4
405 4 -1.0 (41 -42 45 -46 26) #365 u=45 imp:n=1 $ FA #5
406 4 -1.0 (41 -42 45 -46 26) #366 u=46 imp:n=1 $ FA #6
407 4 -1.0 (41 -42 45 -46 26) #367 u=47 imp:n=1 $ FA #7
c
c Guide Channel side & 1 in bottom plate (mixture of H2O & SS304)
41 13 -5.51326 #40 u=4 imp:n=1
412 13 -5.51326 #36 #402 u=42 imp:n=1
413 13 -5.51326 #363 #403 u=43 imp:n=1
414 13 -5.51326 #364 #404 u=44 imp:n=1
415 13 -5.51326 #365 #405 u=45 imp:n=1
416 13 -5.51326 #366 #406 u=46 imp:n=1
417 13 -5.51326 #367 #407 u=47 imp:n=1
c
c Define a row of 6 Horizontal poison pins
c First a pin with axial segments (poison, steel, poison, etc)
44 4 -1.0 -61 -301 u=5 imp:n=1 $ water below bottom B4C
45 10 -1.2225 -60 301 -302 u=5 imp:n=1 $ B4C reduced 75% - Pin 1
46 6 -8.03 60 -61 301 -302 u=5 imp:n=1 $ poison pin SS clad
461 6 -8.03 -61 302 -303 u=5 imp:n=1 $ Steel
462 10 -1.2225 -60 303 -304 u=5 imp:n=1 $ B4C reduced 75% - Pin 2

```


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```
463 6 -8.03 60 -61 303 -304 u=5 imp:n=1 $ poison pin SS clad
464 6 -8.03 -61 304 -305 u=5 imp:n=1 $ Steel
465 10 -1.2225 -60 305 -306 u=5 imp:n=1 $ B4C reduced 75% - Pin 3
466 6 -8.03 60 -61 305 -306 u=5 imp:n=1 $ poison pin SS clad
467 6 -8.03 -61 306 -307 u=5 imp:n=1 $ Steel
468 10 -1.2225 -60 307 -308 u=5 imp:n=1 $ B4C reduced 75% - Pin 4
469 6 -8.03 60 -61 307 -308 u=5 imp:n=1 $ poison pin SS clad
470 6 -8.03 -61 308 -309 u=5 imp:n=1 $ Steel
471 10 -1.2225 -60 309 -310 u=5 imp:n=1 $ B4C reduced 75% - Pin 5
472 6 -8.03 60 -61 309 -310 u=5 imp:n=1 $ poison pin SS clad
473 6 -8.03 -61 310 -311 u=5 imp:n=1 $ Steel
474 10 -1.2225 -60 311 -312 u=5 imp:n=1 $ B4C reduced 75% - Pin 6
475 6 -8.03 60 -61 311 -312 u=5 imp:n=1 $ poison pin SS clad
476 6 -8.03 -61 312 -313 u=5 imp:n=1 $ Steel
477 10 -1.2225 -60 313 -314 u=5 imp:n=1 $ B4C reduced 75% - Pin 7
478 6 -8.03 60 -61 313 -314 u=5 imp:n=1 $ poison pin SS clad
479 6 -8.03 -61 314 -315 u=5 imp:n=1 $ Steel
480 10 -1.2225 -60 315 -316 u=5 imp:n=1 $ B4C reduced 75% - Pin 8
481 6 -8.03 60 -61 315 -316 u=5 imp:n=1 $ poison pin SS clad
482 6 -8.03 -61 316 -317 u=5 imp:n=1 $ Steel
483 10 -1.2225 -60 317 u=5 imp:n=1 $ B4C reduced 75% - Pin 9
484 6 -8.03 60 -61 317 u=5 imp:n=1 $ poison pin SS clad
c
49 4 -1.0 61 u=5 imp:n=1 $ water around pins
c Fill the lattice
60 0 76 -77 72 -73 lat=1 fill= -2:3 0:0 0:0
5 5 5 5 5 u=55 imp:n=1
c Define the horizontal row
61 0 48 -49 72 -73 30 -31 fill=55 trcl=20 imp:n=1 $ Horiz Above FA#1
c Define the rest of the horizontal rows
62 like 61 but trcl=21 imp:n=1 $ Horiz below FA#2
63 like 61 but trcl=22 imp:n=1 $ Horiz below FA#3
64 like 61 but trcl=23 imp:n=1 $ Horiz above FA#4
65 like 61 but trcl=24 imp:n=1 $ Horiz above FA#5
66 like 61 but trcl=25 imp:n=1 $ Horiz below FA#4
67 like 61 but trcl=26 imp:n=1 $ Horiz below FA#1
68 like 61 but trcl=27 imp:n=1 $ Horiz below FA#5
69 like 61 but trcl=28 imp:n=1 $ Horiz above FA#6
70 like 61 but trcl=29 imp:n=1 $ Horiz above FA#7
c
c Define a column of 6 Vertical poison pins
c First a pin with axial segments (poison, steel, poison, etc)
74 4 -1.0 -63 -301 u=6 imp:n=1 $ water below bottom B4C
75 10 -1.2225 -62 301 -302 u=6 imp:n=1 $ B4C reduced 75% - Pin 1
76 6 -8.03 62 -63 301 -302 u=6 imp:n=1 $ poison pin SS clad
761 6 -8.03 -63 302 -303 u=6 imp:n=1 $ Steel
762 10 -1.2225 -62 303 -304 u=6 imp:n=1 $ B4C reduced 75% - Pin 2
763 6 -8.03 62 -63 303 -304 u=6 imp:n=1 $ poison pin SS clad
764 6 -8.03 -63 304 -305 u=6 imp:n=1 $ Steel
765 10 -1.2225 -62 305 -306 u=6 imp:n=1 $ B4C reduced 75% - Pin 3
766 6 -8.03 62 -63 305 -306 u=6 imp:n=1 $ poison pin SS clad
767 6 -8.03 -63 306 -307 u=6 imp:n=1 $ Steel
768 10 -1.2225 -62 307 -308 u=6 imp:n=1 $ B4C reduced 75% - Pin 4
769 6 -8.03 62 -63 307 -308 u=6 imp:n=1 $ poison pin SS clad
770 6 -8.03 -63 308 -309 u=6 imp:n=1 $ Steel
771 10 -1.2225 -62 309 -310 u=6 imp:n=1 $ B4C reduced 75% - Pin 5
772 6 -8.03 62 -63 309 -310 u=6 imp:n=1 $ poison pin SS clad
773 6 -8.03 -63 310 -311 u=6 imp:n=1 $ Steel
774 10 -1.2225 -62 311 -312 u=6 imp:n=1 $ B4C reduced 75% - Pin 6
775 6 -8.03 62 -63 311 -312 u=6 imp:n=1 $ poison pin SS clad
776 6 -8.03 -63 312 -313 u=6 imp:n=1 $ Steel
777 10 -1.2225 -62 313 -314 u=6 imp:n=1 $ B4C reduced 75% - Pin 7
778 6 -8.03 62 -63 313 -314 u=6 imp:n=1 $ poison pin SS clad
779 6 -8.03 -63 314 -315 u=6 imp:n=1 $ Steel
780 10 -1.2225 -62 315 -316 u=6 imp:n=1 $ B4C reduced 75% - Pin 8
781 6 -8.03 62 -63 315 -316 u=6 imp:n=1 $ poison pin SS clad
782 6 -8.03 -63 316 -317 u=6 imp:n=1 $ Steel
783 10 -1.2225 -62 317 u=6 imp:n=1 $ B4C reduced 75% - Pin 9
784 6 -8.03 62 -63 317 u=6 imp:n=1 $ poison pin SS clad
```

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```
c
79 4 -1.0      63                u=6 imp:n=1 $ water around pins
c   Fill the lattice
80 0 76 -78 79 -791 lat=1 fill=6 u=66   imp:n=1
c   Define the vertical column
81 0 44 -47 74 -75 30 -31 fill=66 trcl=30 imp:n=1   $ Vert Right of FA#2
c   Define the rest of the vertical columns
85 like 81 but trcl=31 imp:n=1 $ vert left of FA#3
86 like 81 but trcl=32 imp:n=1 $ vert right of FA#4
87 like 81 but trcl=33 imp:n=1 $ vert left of FA#1
88 like 81 but trcl=34 imp:n=1 $ vert right of FA#1
89 like 81 but trcl=35 imp:n=1 $ vert left of FA#5
90 like 81 but trcl=36 imp:n=1 $ vert right of FA#6
91 like 81 but trcl=37 imp:n=1 $ vert left of FA#7
c
c   Make a dummy water FA in the center
c
42 0 40 -43 44 -47 27 -33 fill=4 imp:n=1
c
c   Define the 6 Outer FAs
c   Make a unit out of the FA, water gap, and guide channel
c
100 0 40 -43 44 -47 27 -33 fill=42 trcl=121 imp:n=1 $ FA #2
101 0 40 -43 44 -47 27 -33 fill=43 trcl=122 imp:n=1 $ FA #3
102 0 40 -43 44 -47 27 -33 fill=44 trcl=123 imp:n=1 $ FA #4
103 0 40 -43 44 -47 27 -33 fill=45 trcl=124 imp:n=1 $ FA #5
104 0 40 -43 44 -47 27 -33 fill=46 trcl=125 imp:n=1 $ FA #6
105 0 40 -43 44 -47 27 -33 fill=47 trcl=126 imp:n=1 $ FA #7
c   Define the 4 basket support rods
106 6 -8.03 -80 27 -33          imp:n=1 $ Upper left
107 6 -8.03 -81 27 -33          imp:n=1 $ Upper right
108 6 -8.03 -82 27 -33          imp:n=1 $ Lower left
109 6 -8.03 -83 27 -33          imp:n=1 $ Lower right
c
c   Define Cask Cells
c
200 6 -8.03 27 -203 100 -101 imp:n=1 $ 317 SS Inner Shell - Radial
201 6 -8.03 202 -27 -101 imp:n=1 $ 304 SS Inner Shell - Bottom
202 6 -8.03 203 -204 -101 imp:n=1 $ 304 SS Inner Shell - Top
203 11 -18.82 (-102 201 -202):(101 -102 202 -204):(-102 204 -205)
      imp:n=1 $ Lead (union of bottom/side/top)
204 6 -8.03 (-103 200 -201):(102 -103 201 -205):(-103 205 -206)
      imp:n=1 $ 304 SST outer shell (union of bottom/side/top)
c
c   Water outside FAs in basket and inside Cask.
c
1000 4 -1.0      -100 27 -203 51 #100 #101 #62 #63 #81 #85 #106 #107
      imp:n=1 $ FAs 2&3 Surrounded by water
1001 4 -1.00    -100 27 -203 50 -51 #42 #102 #103
      #61 #64 #65 #66 #67 #68 #86 #87 #88 #89
      imp:n=1 $ FAs 1,4,5 Surrounded by water
1002 4 -1.00    -100 27 -203 -50 #104 #105 #69 #70 #90 #91 #108 #109
      imp:n=1 $ FAs Surrounded by water
c
c   1st 5 cm water reflector
c
1999 4 -1.00 (206:-200:103) (-207 -104 199) imp:n=1
c
c   Remaining water reflector to 12 in
c
2000 4 -1.0 (207:-199:104) (-208 -105 198) imp:n=0.25
c
c   Outside Cask.
c
3000 0 -1000 (208:-198:105) imp:n=0
c
c   Fuel Cell
c
```

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1 cz .45339 \$ Fuel Pellet OD
2 cz .46228 \$ Zr4 Clad ID
3 cz .53848 \$ Zr4 Clad OD
4 px -0.71501 \$ fuel rod cell boundary
5 px 0.71501 \$ fuel rod cell boundary
6 py -0.71501 \$ fuel rod cell boundary
7 py 0.71501 \$ fuel rod cell boundary
c
c Instrument Tube/Control Rod Guide Tube
c
10 cz 0.64897 \$ ID
11 cz 0.69088 \$ OD
c
c Fuel Assembly
c
12 px -10.7137 \$ 1/2 FA pitch
13 px 10.7137 \$ 1/2 FA pitch
14 py -10.7137 \$ 1/2 FA pitch
15 py 10.7137 \$ 1/2 FA pitch
c
c Zr grid inner pitch
c
16 px -0.712475 \$ Zr grid cell boundary
17 px 0.712475 \$ Zr grid cell boundary
18 py -0.712475 \$ Zr grid cell boundary
19 py 0.712475 \$ Zr grid cell boundary
c
c Overall Fuel Height including blanket - 144 inches
c
20 pz -182.88
21 pz 182.88
c
c Axial Blanket/enriched fuel zone - 6 inches below top & bottom
c
22 pz -167.64
23 pz 167.64
c
c Top and bottom plenum regions
c
24 pz -185.75
25 pz 204.673
c
c Channel bottom plate
c
26 pz -192.151
27 pz -194.691
c
c Lower Tie Plate - use FA px/py (8.424 in pitch vs 8.436 in for FA)
c
c Smear mass over rectangle for now
c Bottom of LTP
30 pz -186.953 \$ Water to bottom of lowest poison pin B4C
c Surfaces for poison pin/steel segments
301 pz -178.65725 \$ Poison in lowest poison pin to steel
302 pz -138.85545 \$ Steel to bottom of next poison pin B4C
303 pz -125.8062 \$ Poison in next poison pin to steel
304 pz -91.8718 \$ Steel to bottom of next poison pin B4C
305 pz -78.82255 \$ Poison in next poison pin to steel
306 pz -44.88815 \$ Steel to bottom of next poison pin B4C
307 pz -31.8389 \$ Poison in next poison pin to steel
308 pz 2.0955 \$ Steel to bottom of next poison pin B4C
309 pz 15.14475 \$ Poison in next poison pin to steel
310 pz 49.07915 \$ Steel to bottom of next poison pin B4C
311 pz 62.1284 \$ Poison in next poison pin to steel
312 pz 96.0628 \$ Steel to bottom of next poison pin B4C
313 pz 109.11205 \$ Poison in next poison pin to steel
314 pz 143.04645 \$ Steel to bottom of next poison pin B4C
315 pz 156.0957 \$ Poison in next poison pin to steel
316 pz 190.0301 \$ Steel to bottom of top poison pin B4C

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317 pz 203.07935 $ Poison in top poison pin to steel - through UTP
c
c Upper Tie Plate - use FA px/py (8.424 in pitch vs 8.436 in for FA)
c
c Smear mass over rectangle for now
c Top of UTP
31 pz 206.538
c
c pz 213.512 $ Top of FA
c
c Square Cell for Channel
c
40 px -11.3816
41 px -11.1919
42 px 11.1919
43 px 11.3816
44 py -11.3816
45 py -11.1919
46 py 11.1919
47 py 11.3816
c
c Rectangular Cell for Basket (ignore 9 - 1" thick axial spacers)
c
48 px -11.43
49 px 11.43
50 py -13.0548
51 py 13.0548
c
c B4C poison pins -
60 c/z 1.905 0.8763 0.5842 $ ID of SS304 clad
61 c/z 1.905 0.8763 0.635 $ OD of SS304 clad
62 c/z 1.1049 1.905 0.5842 $ ID of SS304 clad
63 c/z 1.1049 1.905 0.635 $ OD of SS304 clad
72 py 0.0397 $
73 py 1.7129 $ 1/2 pitch for horiz poison pin row (1/2 of 1.38 in)
74 px 0.0397 $
75 px 2.3352 $ 1/2 pitch for vertic poison pin row (1/2 of 1.87 in)
76 px 0.0 $ 1/2 x dimension pitch for poison pin row
77 px 3.81
78 px 2.2098
79 py 0.0
791 py 3.81
c
c Four Basket Support Rods - 2-1/4 inch 216 SS
c
80 c/z -29.21 26.43124 2.8575
81 c/z 29.21 26.43124 2.8575
82 c/z -29.21 -26.43124 2.8575
83 c/z 29.21 -26.43124 2.8575
c
c Cask dimensions
c
c Radial
100 cz 47.625 $ Inner Cavity -18.75"
101 cz 48.895 $ 317 SS inner - 0.5"
102 cz 59.055 $ Cast DU - 4"
103 cz 63.0174 $ 317 SS outer -1.56"
104 cz 68.0174 $ 5 cm side reflector
105 cz 93.4974 $ 12 inch side reflector
c Axial - Bottom to Top of Cask
198 pz -241.6810 $ 12 inch bottom reflector
199 pz -216.2010 $ 5 cm bottom reflector
200 pz -211.201 $ 304 SST - 1.5"
201 pz -207.391 $ Cast DU - 3.75"
202 pz -197.866 $ 304 SST - 1.25"
203 pz 235.839 $ Cavity -169.5"
204 pz 238.379 $ 304 SST - 1"
205 pz 245.999 $ Cast DU - 3"
206 pz 251.079 $ 304 SST - 2"

```

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207 pz 256.0790 $ 5 cm top reflector
208 pz 281.5590 $ 12 inch top reflector
c
c Axial planes for spacer grid
c
450 pz 199.3265 $ Top of top spacer
451 pz 194.8815
452 pz 152.0063
453 pz 147.5613
454 pz 117.4115
455 pz 115.5065
456 pz 85.4837
457 pz 81.0387
458 pz 50.8889
459 pz 48.9839
460 pz 18.9611
461 pz 14.5161
462 pz -15.6337
463 pz -17.5387
464 pz -47.5615
465 pz -52.0065
466 pz -114.0841
467 pz -118.5291
468 pz -174.9552
469 pz -180.6702 $ 1st spacer from bottom
1000 so 1000

c Translations for horizontal row of 6 poison pins
tr1 -9.525 0.8763 0 $ Left
tr2 -5.715 0.8763 0 $
tr3 -1.905 0.8763 0 $
tr4 1.905 0.8763 0 $
tr5 5.715 0.8763 0 $
tr6 9.525 0.8763 0 $ Right
c Translations for vertical row of 6 poison pins
tr11 1.18745 -9.525 0 $ Bottom
tr12 1.18745 -5.715 0 $
tr13 1.18745 -1.905 0 $
tr14 1.18745 1.905 0 $
tr15 1.18745 5.715 0 $
tr16 1.18745 9.525 0 $ Top
c Translate horizontal poison pin rows
tr20 0.0000 11.3419 0 $ Horiz above FA#1
tr21 -13.6771 13.0151 0 $ Horiz below FA#2
tr22 13.6771 13.0151 0 $ Horiz below FA#3
tr23 -27.3542 11.3419 0 $ Horiz above FA#4
tr24 27.3542 11.3419 0 $ Horiz above FA#5
tr25 -27.3542 -13.0945 0 $ Horiz below FA#4
tr26 0.0000 -13.0945 0 $ Horiz below FA#1
tr27 27.3542 -13.0945 0 $ Horiz below FA#5
tr28 -13.6771 -14.7677 0 $ Horiz above FA#6
tr29 13.6771 -14.7677 0 $ Horiz above FA#7
c Translate vertical poison pin rows
tr30 -2.3352 26.1096 0 $ vert right of FA#2
tr31 -0.0397 26.1096 0 $ vert left of FA#3
tr32 -16.0123 0.0000 0 $ vert right of FA#4
tr33 -13.7168 0.0000 0 $ vert left of FA#1
tr34 11.3419 0.0000 0 $ vert right of FA#1
tr35 13.6374 0.0000 0 $ vert left of FA#5
tr36 -2.3352 -26.1096 0 $ vert right of FA#6
tr37 -0.0397 -26.1096 0 $ vert left of FA#7
c FA translations
tr121 -13.6771 26.1096 0 $ FA #2
tr122 13.6771 26.1096 0 $ FA #3
tr123 -27.3542 0 0 $ FA #4
tr124 27.3542 0 0 $ FA #5
tr125 -13.6771 -26.1096 0 $ FA #6
tr126 13.6771 -26.1096 0 $ FA #7
c Shift FA within basket channel

```

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```

trl32 -0.478 0.478 0 $ FA #2
trl33 0.478 0.478 0 $ FA #3
trl34 -0.478 0 0 $ FA #4
trl35 0.478 0 0 $ FA #5
trl36 -0.478 -0.478 0 $ FA #6
trl37 0.478 -0.478 0 $ FA #7
mode n
kcode 3000 1 30 220 50000
print -128
prdmp j -120 j 3
fc4 fission * lethargy**2 in cell 1
f4:n 1
fm4 -1 1 -6
sd4 1.0
c lethargy**2 [base 20 MeV -- u=0 at 20 MeV]
de4 1.e-8 1.e-7 1.e-6 1.e-5 1.e-4 .001 .01 .1 1 10 14
df4 458.7 365.3 282.6 210.5 149.0 98.08 57.77 28.07 8.974 0.48 .127
fc14 fission in cell 1
f14:n 1
fm14 -1 1 -6
sd14 1.0
fc24 fission * lethargy in cell 1
f24:n 1
fm24 -1 1 -6
sd24 1.0
c lethargy [base 20 MeV -- u=0 at 20 MeV]
de24 1.e-8 1.e-7 1.e-6 1.e-5 1.e-4 .001 .01 .1 1 10 14
df24 21.42 19.11 16.81 14.51 12.21 9.90 7.60 5.298 2.996 0.693 0.357
c m1 is UO2 fuel - Enriched to 4.25 wt% U235
m1 92235.50c -0.037462
92238.50c -0.843989
8016.50c -0.118549
c m2 is He
m2 2004.50c -1.0
c m3 is Zircaloy 4
m3 40000.60c -98.18
8016.50c -0.12
24000.50c -0.10
26000.55c -0.20
50000.35c -1.40
c m4 is water
m4 1001.50c 0.666700 8016.50c 0.333300
c m5 is Natural UO2 blanket fuel - 0.72 wt% U235
m5 92235.50c -0.0062672
92238.50c -0.875206
8016.50c -0.118528
c m6 is 304L SS
m6 6000.50c -0.03
14000.50c -1
15031.50c -0.045
16032.50c -0.03
24000.50c -20
25055.50c -2
26000.55c -64.895
28000.50c -12
c m7 is Inconel 718
m7 5000.01c -0.006
6000.50c -0.08
13027.50c -0.8
14000.50c -0.35
22000.50c -1.15
24000.50c -21
25055.50c -0.35
26000.55c -11.164
27059.50c -1
28000.50c -55
29000.50c -0.3
41093.50c -5.5
42000.50c -3.3

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c m8 is Inconel X750
m8 6000.50c -0.08
13027.50c -1
14000.50c -0.5
22000.50c -2.75
24000.50c -15
25055.50c -1
26000.55c -7
27059.50c -1
28000.50c -70
29000.50c -0.5
41093.50c -1.2
c m10 is B4C poison
m10 5010.50c -0.140886 \$ B10 lower end 18.3% - 0.3%
5011.50c -0.641812 \$ B11 upper end 81.7% + 0.3%
6000.50c -0.217302 \$ C
c m11 is Depleted Uranium
m11 92235.50c 0.000106128
92238.50c 0.0475275
c m12 is Air
m12 7014.50c -0.765
8016.50c -0.235
c m13 is Channel - 304 SS and water mixture (VF water = 0.358)
m13 6000.50c -0.000748
14000.50c -0.009351
15031.50c -0.000421
16032.50c -0.000281
24000.50c -0.187013
25055.50c -0.018701
26000.55c -0.620369
28000.50c -0.098182
1001.50c -0.007266
8016.50c -0.057669
mt4 lwtr.01t
mt13 lwtr.01t

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Case pso6i10z

PWR 6 FAs 4.25%, CTMT/INT- 1.0/0.0 -Inf Hex- 2 cm - Shifted Out - Inf T/B Refl

```

1 1 -10.406 -1 22 -23 u=1 imp:n=1 $ Enriched uo2 fuel
2 4 -1.0 1 -2 22 -23 u=1 imp:n=1 $ gap - H2O
3 3 -6.565 2 -3 22 -23 u=1 imp:n=1 $ Zr4 Clad
4 4 -1.0 3 16 -17 18 -19 22 -23 u=1 imp:n=1 $ Moderator
501 4 -1.0 (-16:17:-18:19) 22 -467 u=1 imp:n=1 $ Spacer - Moderator
502 3 -6.565 (-16:17:-18:19) 467 -466 u=1 imp:n=1 $ Zr4 - 2nd Spacer
503 4 -1.0 (-16:17:-18:19) 466 -465 u=1 imp:n=1 $ Spacer - Moderator
504 3 -6.565 (-16:17:-18:19) 465 -464 u=1 imp:n=1 $ Zr4 - 3rd Spacer
505 4 -1.0 (-16:17:-18:19) 464 -463 u=1 imp:n=1 $ Spacer - Moderator
506 3 -6.565 (-16:17:-18:19) 463 -462 u=1 imp:n=1 $ Zr4 - 4th Spacer
507 4 -1.0 (-16:17:-18:19) 462 -461 u=1 imp:n=1 $ Spacer - Moderator
508 3 -6.565 (-16:17:-18:19) 461 -460 u=1 imp:n=1 $ Zr4 - 5th Spacer
509 4 -1.0 (-16:17:-18:19) 460 -459 u=1 imp:n=1 $ Spacer - Moderator
510 3 -6.565 (-16:17:-18:19) 459 -458 u=1 imp:n=1 $ Zr4 - 6th Spacer
511 4 -1.0 (-16:17:-18:19) 458 -457 u=1 imp:n=1 $ Spacer - Moderator
512 3 -6.565 (-16:17:-18:19) 457 -456 u=1 imp:n=1 $ Zr4 - 7th Spacer
513 4 -1.0 (-16:17:-18:19) 456 -455 u=1 imp:n=1 $ Spacer - Moderator
514 3 -6.565 (-16:17:-18:19) 455 -454 u=1 imp:n=1 $ Zr4 - 8th Spacer
515 4 -1.0 (-16:17:-18:19) 454 -453 u=1 imp:n=1 $ Spacer - Moderator
516 3 -6.565 (-16:17:-18:19) 453 -452 u=1 imp:n=1 $ Zr4 - 9th Spacer
517 4 -1.0 (-16:17:-18:19) 452 -23 u=1 imp:n=1 $ Spacer - Moderator
c Natural Uranium blanket - Top
5 5 -10.406 -1 23 -21 u=1 imp:n=1 $ Natural uo2 fuel
6 4 -1.0 1 -2 23 -21 u=1 imp:n=1 $ gap - H2O
7 3 -6.565 2 -3 23 -21 u=1 imp:n=1 $ Zr4 Clad
8 4 -1.0 3 23 -21 u=1 imp:n=1 $ Moderator
c Natural Uranium blanket - Bottom
9 5 -10.406 -1 20 -22 u=1 imp:n=1 $ Natural uo2 fuel
10 4 -1.0 1 -2 20 -22 u=1 imp:n=1 $ gap - H2O
11 3 -6.565 2 -3 20 -22 u=1 imp:n=1 $ Zr4 Clad
12 4 -1.0 3 20 -22 u=1 imp:n=1 $ Moderator
c Plenum region - below fuel pins
13 4 -1.0 -2 24 -20 u=1 imp:n=1 $ He
14 3 -6.565 2 -3 24 -20 u=1 imp:n=1 $ Zr4 Clad
15 4 -1.0 3 24 -20 u=1 imp:n=1 $ Moderator
c Plenum region - above fuel pins
16 4 -1.0 -2 21 -25 u=1 imp:n=1 $ He
17 3 -6.565 2 -3 21 -25 u=1 imp:n=1 $ Zr4 Clad
18 4 -1.0 3 21 -25 u=1 imp:n=1 $ Moderator
c Rest of Lower tie plate - below fuel pins
19 6 -8.03 30 -24 u=1 imp:n=1 $ 304L SS in LTP
c Water below Lower tie plate/above channel bottom -below fuel pins
191 4 -1.0 -30 u=1 imp:n=1 $ Water
c Rest of Upper tie plate - above fuel pins
20 6 -8.03 25 u=1 imp:n=1 $ 304L SS in LTP
c Lattice of Cells - Top of water above UTP to water below LTP
25 0 4 -5 6 -7 lat=1 u=2 fill= -7:7 -7:7 0:0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ x=-7 to x=7:y=7
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=6
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1 1 $ " ";y=5
1 1 1 1 1 1 1 1 3 1 1 1 1 1 1 1 $ " ";y=4
1 1 1 1 3 1 1 1 1 1 3 1 1 1 1 1 $ " ";y=3
1 1 3 1 1 1 1 1 1 1 1 1 3 1 1 1 $ " ";y=2
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=1
1 1 1 3 1 1 1 1 3 1 1 1 3 1 1 1 $ " ";y=0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=-1
1 1 3 1 1 1 1 1 1 1 1 1 3 1 1 1 $ " ";y=-2
1 1 1 1 3 1 1 1 1 1 3 1 1 1 1 1 $ " ";y=-3
1 1 1 1 1 1 1 1 3 1 1 1 1 1 1 1 $ " ";y=-4
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1 1 $ " ";y=-5
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ " ";y=-6
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 imp:n=1 $ " ";y=-7
c Instrument/Control Rod Guide Tubes
26 4 -1.0 -10 24 -25 u=3 imp:n=1 $ water inside tube
27 3 -6.565 10 -11 24 -25 u=3 imp:n=1 $ Zr4 Clad

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```

28  4 -1.0  11 16 -17 18 -19  24 -25 u=3 imp:n=1 $ Moderator
281 4 -1.0  (-16:17:-18:19) 24 -467 u=3 imp:n=1 $ Spacer - Moderator
282 3 -6.565 (-16:17:-18:19) 467 -466 u=3 imp:n=1 $ Zr4 - 2nd Spacer
283 4 -1.0  (-16:17:-18:19) 466 -465 u=3 imp:n=1 $ Spacer - Moderator
284 3 -6.565 (-16:17:-18:19) 465 -464 u=3 imp:n=1 $ Zr4 - 3rd Spacer
285 4 -1.0  (-16:17:-18:19) 464 -463 u=3 imp:n=1 $ Spacer - Moderator
286 3 -6.565 (-16:17:-18:19) 463 -462 u=3 imp:n=1 $ Zr4 - 4th Spacer
287 4 -1.0  (-16:17:-18:19) 462 -461 u=3 imp:n=1 $ Spacer - Moderator
288 3 -6.565 (-16:17:-18:19) 461 -460 u=3 imp:n=1 $ Zr4 - 5th Spacer
289 4 -1.0  (-16:17:-18:19) 460 -459 u=3 imp:n=1 $ Spacer - Moderator
290 3 -6.565 (-16:17:-18:19) 459 -458 u=3 imp:n=1 $ Zr4 - 6th Spacer
291 4 -1.0  (-16:17:-18:19) 458 -457 u=3 imp:n=1 $ Spacer - Moderator
292 3 -6.565 (-16:17:-18:19) 457 -456 u=3 imp:n=1 $ Zr4 - 7th Spacer
293 4 -1.0  (-16:17:-18:19) 456 -455 u=3 imp:n=1 $ Spacer - Moderator
294 3 -6.565 (-16:17:-18:19) 455 -454 u=3 imp:n=1 $ Zr4 - 8th Spacer
295 4 -1.0  (-16:17:-18:19) 454 -453 u=3 imp:n=1 $ Spacer - Moderator
296 3 -6.565 (-16:17:-18:19) 453 -452 u=3 imp:n=1 $ Zr4 - 9th Spacer
297 4 -1.0  (-16:17:-18:19) 452 -25 u=3 imp:n=1 $ Spacer - Moderator
c    Lower Tie Plate
c    Instrument/Control Rod Guide Tube Locations
29  4 -1.0  -10      -24 30      u=3 imp:n=1 $ water inside tube
30  3 -6.565 10 -11      -24 30      u=3 imp:n=1 $ Zr4 Clad
31  6 -8.03 11      -24 30      u=3 imp:n=1 $ 304L SS in LTP
c    Water below Lower tie plate/above channel bottom -below fuel pins
311 4 -1.0  -30      u=3 imp:n=1 $ Water
c    Upper Tie Plate
c    Instrument/Control Rod Guide Tube Locations
33  4 -1.0  -10      25      u=3 imp:n=1 $ water inside tube
34  3 -6.565 10 -11 25      u=3 imp:n=1 $ Zr4 Clad
35  6 -8.03 11      25      u=3 imp:n=1 $ 304L SS in UTP
c    Fuel Assembly
36  0 12 -13 14 -15 26 -31 u=42 fill=2 trcl=132 imp:n=1 $ FA #2
363 0 12 -13 14 -15 26 -31 u=43 fill=2 trcl=133 imp:n=1 $ FA #3
364 0 12 -13 14 -15 26 -31 u=44 fill=2 trcl=134 imp:n=1 $ FA #4
365 0 12 -13 14 -15 26 -31 u=45 fill=2 trcl=135 imp:n=1 $ FA #5
366 0 12 -13 14 -15 26 -31 u=46 fill=2 trcl=136 imp:n=1 $ FA #6
367 0 12 -13 14 -15 26 -31 u=47 fill=2 trcl=137 imp:n=1 $ FA #7
c
c    Water Gap between fuel assembly and guide channel
c    and above UTP.
c
40  4 -1.0  (41 -42 45 -46 26)      u=4  imp:n=1  $ FA #1 - Center Hole
402 4 -1.0  (41 -42 45 -46 26) #36 u=42 imp:n=1  $ FA #2
403 4 -1.0  (41 -42 45 -46 26) #363 u=43 imp:n=1  $ FA #3
404 4 -1.0  (41 -42 45 -46 26) #364 u=44 imp:n=1  $ FA #4
405 4 -1.0  (41 -42 45 -46 26) #365 u=45 imp:n=1  $ FA #5
406 4 -1.0  (41 -42 45 -46 26) #366 u=46 imp:n=1  $ FA #6
407 4 -1.0  (41 -42 45 -46 26) #367 u=47 imp:n=1  $ FA #7
c
c    Guide Channel side & 1 in bottom plate (mixture of H2O & SS304)
41  13 -5.51326 #40 u=4  imp:n=1
412 13 -5.51326 #36 #402 u=42 imp:n=1
413 13 -5.51326 #363 #403 u=43 imp:n=1
414 13 -5.51326 #364 #404 u=44 imp:n=1
415 13 -5.51326 #365 #405 u=45 imp:n=1
416 13 -5.51326 #366 #406 u=46 imp:n=1
417 13 -5.51326 #367 #407 u=47 imp:n=1
c
c    Define a row of 6 Horizontal poison pins
c    First a pin with axial segments (poison, steel, poison, etc)
44  4 -1.0  -61      -301      u=5 imp:n=1 $ water below bottom B4C
45  10 -1.2225 -60 301 -302      u=5 imp:n=1 $ B4C reduced 75% - Pin 1
46  6 -8.03  60 -61 301 -302      u=5 imp:n=1 $ poison pin SS clad
461 6 -8.03  -61 302 -303      u=5 imp:n=1 $ Steel
462 10 -1.2225 -60 303 -304      u=5 imp:n=1 $ B4C reduced 75% - Pin 2
463 6 -8.03  60 -61 303 -304      u=5 imp:n=1 $ poison pin SS clad
464 6 -8.03  -61 304 -305      u=5 imp:n=1 $ Steel
465 10 -1.2225 -60 305 -306      u=5 imp:n=1 $ B4C reduced 75% - Pin 3
466 6 -8.03  60 -61 305 -306      u=5 imp:n=1 $ poison pin SS clad

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467 6 -8.03      -61 306 -307      u=5 imp:n=1 $ Steel
468 10 -1.2225   -60 307 -308      u=5 imp:n=1 $ B4C reduced 75% - Pin 4
469 6 -8.03 60   -61 307 -308      u=5 imp:n=1 $ poison pin SS clad
470 6 -8.03      -61 308 -309      u=5 imp:n=1 $ Steel
471 10 -1.2225   -60 309 -310      u=5 imp:n=1 $ B4C reduced 75% - Pin 5
472 6 -8.03 60   -61 309 -310      u=5 imp:n=1 $ poison pin SS clad
473 6 -8.03      -61 310 -311      u=5 imp:n=1 $ Steel
474 10 -1.2225   -60 311 -312      u=5 imp:n=1 $ B4C reduced 75% - Pin 6
475 6 -8.03 60   -61 311 -312      u=5 imp:n=1 $ poison pin SS clad
476 6 -8.03      -61 312 -313      u=5 imp:n=1 $ Steel
477 10 -1.2225   -60 313 -314      u=5 imp:n=1 $ B4C reduced 75% - Pin 7
478 6 -8.03 60   -61 313 -314      u=5 imp:n=1 $ poison pin SS clad
479 6 -8.03      -61 314 -315      u=5 imp:n=1 $ Steel
480 10 -1.2225   -60 315 -316      u=5 imp:n=1 $ B4C reduced 75% - Pin 8
481 6 -8.03 60   -61 315 -316      u=5 imp:n=1 $ poison pin SS clad
482 6 -8.03      -61 316 -317      u=5 imp:n=1 $ Steel
483 10 -1.2225   -60 317
484 6 -8.03 60   -61 317      u=5 imp:n=1 $ poison pin SS clad
c
49  4 -1.0      61                u=5 imp:n=1 $ water around pins
c    Fill the lattice
60  0 76 -77 72 -73 lat=1 fill= -2:3 0:0 0:0
      5 5 5 5 5 u=55 imp:n=1
c    Define the horizontal row
61  0 48 -49 72 -73 30 -31 fill=55 trcl=20 imp:n=1 $ Horiz Above FA#1
c    Define the rest of the horizontal rows
62  like 61 but trcl=21 imp:n=1 $ Horiz below FA#2
63  like 61 but trcl=22 imp:n=1 $ Horiz below FA#3
64  like 61 but trcl=23 imp:n=1 $ Horiz above FA#4
65  like 61 but trcl=24 imp:n=1 $ Horiz above FA#5
66  like 61 but trcl=25 imp:n=1 $ Horiz below FA#4
67  like 61 but trcl=26 imp:n=1 $ Horiz below FA#1
68  like 61 but trcl=27 imp:n=1 $ Horiz below FA#5
69  like 61 but trcl=28 imp:n=1 $ Horiz above FA#6
70  like 61 but trcl=29 imp:n=1 $ Horiz above FA#7
c
c    Define a column of 6 Vertical poison pins
c    First a pin with axial segments (poison, steel, poison, etc)
74  4 -1.0      -63      -301      u=6 imp:n=1 $ water below bottom B4C
75  10 -1.2225   -62 301 -302      u=6 imp:n=1 $ B4C reduced 75% - Pin 1
76  6 -8.03 62   -63 301 -302      u=6 imp:n=1 $ poison pin SS clad
761 6 -8.03      -63 302 -303      u=6 imp:n=1 $ Steel
762 10 -1.2225   -62 303 -304      u=6 imp:n=1 $ B4C reduced 75% - Pin 2
763 6 -8.03 62   -63 303 -304      u=6 imp:n=1 $ poison pin SS clad
764 6 -8.03      -63 304 -305      u=6 imp:n=1 $ Steel
765 10 -1.2225   -62 305 -306      u=6 imp:n=1 $ B4C reduced 75% - Pin 3
766 6 -8.03 62   -63 305 -306      u=6 imp:n=1 $ poison pin SS clad
767 6 -8.03      -63 306 -307      u=6 imp:n=1 $ Steel
768 10 -1.2225   -62 307 -308      u=6 imp:n=1 $ B4C reduced 75% - Pin 4
769 6 -8.03 62   -63 307 -308      u=6 imp:n=1 $ poison pin SS clad
770 6 -8.03      -63 308 -309      u=6 imp:n=1 $ Steel
771 10 -1.2225   -62 309 -310      u=6 imp:n=1 $ B4C reduced 75% - Pin 5
772 6 -8.03 62   -63 309 -310      u=6 imp:n=1 $ poison pin SS clad
773 6 -8.03      -63 310 -311      u=6 imp:n=1 $ Steel
774 10 -1.2225   -62 311 -312      u=6 imp:n=1 $ B4C reduced 75% - Pin 6
775 6 -8.03 62   -63 311 -312      u=6 imp:n=1 $ poison pin SS clad
776 6 -8.03      -63 312 -313      u=6 imp:n=1 $ Steel
777 10 -1.2225   -62 313 -314      u=6 imp:n=1 $ B4C reduced 75% - Pin 7
778 6 -8.03 62   -63 313 -314      u=6 imp:n=1 $ poison pin SS clad
779 6 -8.03      -63 314 -315      u=6 imp:n=1 $ Steel
780 10 -1.2225   -62 315 -316      u=6 imp:n=1 $ B4C reduced 75% - Pin 8
781 6 -8.03 62   -63 315 -316      u=6 imp:n=1 $ poison pin SS clad
782 6 -8.03      -63 316 -317      u=6 imp:n=1 $ Steel
783 10 -1.2225   -62 317
784 6 -8.03 62   -63 317      u=6 imp:n=1 $ poison pin SS clad
c
79  4 -1.0      63                u=6 imp:n=1 $ water around pins
c    Fill the lattice
80  0 76 -78 79 -791 lat=1 fill=6 u=66 imp:n=1

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c   Define the vertical column
81  0 44 -47 74 -75 30 -31 fill=66 trcl=30 imp:n=1  $ Vert Right of FA#2
c   Define the rest of the vertical columns
85  like 81 but trcl=31 imp:n=1 $ vert left of FA#3
86  like 81 but trcl=32 imp:n=1 $ vert right of FA#4
87  like 81 but trcl=33 imp:n=1 $ vert left of FA#1
88  like 81 but trcl=34 imp:n=1 $ vert right of FA#1
89  like 81 but trcl=35 imp:n=1 $ vert left of FA#5
90  like 81 but trcl=36 imp:n=1 $ vert right of FA#6
91  like 81 but trcl=37 imp:n=1 $ vert left of FA#7
c
c   Make a dummy water FA in the center
c
42  0 40 -43 44 -47 27 -33 fill=4 imp:n=1
c
c   Define the 6 Outer FAs
c   Make a unit out of the FA, water gap, and guide channel
c
100  0 40 -43 44 -47 27 -33 fill=42 trcl=121 imp:n=1 $ FA #2
101  0 40 -43 44 -47 27 -33 fill=43 trcl=122 imp:n=1 $ FA #3
102  0 40 -43 44 -47 27 -33 fill=44 trcl=123 imp:n=1 $ FA #4
103  0 40 -43 44 -47 27 -33 fill=45 trcl=124 imp:n=1 $ FA #5
104  0 40 -43 44 -47 27 -33 fill=46 trcl=125 imp:n=1 $ FA #6
105  0 40 -43 44 -47 27 -33 fill=47 trcl=126 imp:n=1 $ FA #7
c
c   Define the 4 basket support rods
106  6 -8.03 -80 27 -33 imp:n=1 $ Upper left
107  6 -8.03 -81 27 -33 imp:n=1 $ Upper right
108  6 -8.03 -82 27 -33 imp:n=1 $ Lower left
109  6 -8.03 -83 27 -33 imp:n=1 $ Lower right
c
c   Define Cask Cells
c
200  6 -8.03 27 -203 100 -101 imp:n=1 $ 317 SS Inner Shell - Radial
201  6 -8.03 202 -27 -101 imp:n=1 $ 304 SS Inner Shell - Bottom
202  6 -8.03 203 -204 -101 imp:n=1 $ 304 SS Inner Shell - Top
203  11 -18.82 (-102 201 -202):(101 -102 202 -204):(-102 204 -205)
      imp:n=1 $ Lead (union of bottom/side/top)
204  6 -8.03 (-103 200 -201):(102 -103 201 -205):(-103 205 -206)
      imp:n=1 $ 304 SST outer shell (union of bottom/side/top)
c
c   Water outside FAs in basket and inside Cask.
c
1000  4 -1.0 -100 27 -203 51 #100 #101 #62 #63 #81 #85 #106 #107
      imp:n=1 $ FAs 2&3 Surrounded by water
1001  4 -1.0 -100 27 -203 50 -51 #42 #102 #103
      #61 #64 #65 #66 #67 #68 #86 #87 #88 #89
      imp:n=1 $ FAs 1,4,5 Surrounded by water
1002  4 -1.0 -100 27 -203 -50 #104 #105 #69 #70 #90 #91 #108 #109
      imp:n=1 $ FAs Surrounded by water
c
c   1 cm water reflector
c   Water around cask
c
2000  4 -0.0001 (-52 53 -54 55 -56 57 58 -59) (206:-200:103) imp:n=1
c
2001  0 (52:-53:54:-55:56:-57:-58:59) imp:n=0
c
c   Fuel Cell
c
1  cz .45339 $ Fuel Pellet OD
2  cz .46228 $ Zr4 Clad ID
3  cz .53848 $ Zr4 Clad OD
4  px -0.71501 $ fuel rod cell boundary
5  px 0.71501 $ fuel rod cell boundary
6  py -0.71501 $ fuel rod cell boundary
7  py 0.71501 $ fuel rod cell boundary
c

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c Instrument Tube/Control Rod Guide Tube
c
10 cz 0.64897 \$ ID
11 cz 0.69088 \$ OD
c
c Fuel Assembly
c
12 px -10.7137 \$ 1/2 FA pitch
13 px 10.7137 \$ 1/2 FA pitch
14 py -10.7137 \$ 1/2 FA pitch
15 py 10.7137 \$ 1/2 FA pitch
c
c Zr grid inner pitch
c
16 px -0.712475 \$ Zr grid cell boundary
17 px 0.712475 \$ Zr grid cell boundary
18 py -0.712475 \$ Zr grid cell boundary
19 py 0.712475 \$ Zr grid cell boundary
c
c Overall Fuel Height including blanket - 144 inches
c
20 pz -182.88
21 pz 182.88
c
c Axial Blanket/enriched fuel zone - 6 inches below top & bottom
c
22 pz -167.64
23 pz 167.64
c
c Top and bottom plenum regions
c
24 pz -185.75
25 pz 204.673
c
c Channel bottom plate
c
26 pz -192.151
27 pz -194.691
c
c Lower Tie Plate - use FA px/py (8.424 in pitch vs 8.436 in for FA)
c
c Smear mass over rectangle for now
c Bottom of LTP
30 pz -186.953 \$ Water to bottom of lowest poison pin B4C
c Surfaces for poison pin/steel segments
301 pz -178.65725 \$ Poison in lowest poison pin to steel
302 pz -138.85545 \$ Steel to bottom of next poison pin B4C
303 pz -125.8062 \$ Poison in next poison pin to steel
304 pz -91.8718 \$ Steel to bottom of next poison pin B4C
305 pz -78.82255 \$ Poison in next poison pin to steel
306 pz -44.88815 \$ Steel to bottom of next poison pin B4C
307 pz -31.8389 \$ Poison in next poison pin to steel
308 pz 2.0955 \$ Steel to bottom of next poison pin B4C
309 pz 15.14475 \$ Poison in next poison pin to steel
310 pz 49.07915 \$ Steel to bottom of next poison pin B4C
311 pz 62.1284 \$ Poison in next poison pin to steel
312 pz 96.0628 \$ Steel to bottom of next poison pin B4C
313 pz 109.11205 \$ Poison in next poison pin to steel
314 pz 143.04645 \$ Steel to bottom of next poison pin B4C
315 pz 156.0957 \$ Poison in next poison pin to steel
316 pz 190.0301 \$ Steel to bottom of top poison pin B4C
317 pz 203.07935 \$ Poison in top poison pin to steel - through UTP
c
c Upper Tie Plate - use FA px/py (8.424 in pitch vs 8.436 in for FA)
c
c Smear mass over rectangle for now
c Top of UTP
31 pz 206.538
c

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33 pz 213.512 $ Top of FA
c
c Square Cell for Channel
c
40 px -11.3816
41 px -11.1919
42 px 11.1919
43 px 11.3816
44 py -11.3816
45 py -11.1919
46 py 11.1919
47 py 11.3816
c
c Rectangular Cell for Basket (ignore 9 - 1" thick axial spacers)
c
48 px -11.43
49 px 11.43
50 py -13.0548
51 py 13.0548
c
c Hexagonal Cell Surrounding Cask for Infinite array
c
c inner hex surfaces bounding lattice, (n-0.5)*p*cos30
*52 p 0.8660254 -0.5 0.0 64.000
*53 p 0.8660254 -0.5 0.0 -64.000
*54 p 0.8660254 0.5 0.0 64.000
*55 p 0.8660254 0.5 0.0 -64.000
*56 py 64.000
*57 py -64.000
*58 pz -212.201 $ 2 cm bottom pitch
*59 pz 252.079 $ 2 cm top pitch
c
c B4C poison pins -
60 c/z 1.905 0.8763 0.5842 $ ID of SS304 clad
61 c/z 1.905 0.8763 0.635 $ OD of SS304 clad
62 c/z 1.1049 1.905 0.5842 $ ID of SS304 clad
63 c/z 1.1049 1.905 0.635 $ OD of SS304 clad
72 py 0.0397 $
73 py 1.7129 $ 1/2 pitch for horiz poison pin row (1/2 of 1.38 in)
74 px 0.0397 $
75 px 2.3352 $ 1/2 pitch for vertic poison pin row (1/2 of 1.87 in)
76 px 0.0 $ 1/2 x dimension pitch for poison pin row
77 px 3.81
78 px 2.2098
79 py 0.0
791 py 3.81
c
c Four Basket Support Rods - 2-1/4 inch 216 SS
c
80 c/z -29.21 26.43124 2.8575
81 c/z 29.21 26.43124 2.8575
82 c/z -29.21 -26.43124 2.8575
83 c/z 29.21 -26.43124 2.8575
c
c Cask dimensions
c
c Radial
100 cz 47.625 $ Inner Cavity -18.75"
101 cz 48.895 $ 317 SS inner - 0.5"
102 cz 59.055 $ Cast DU - 4"
103 cz 63.0174 $ 317 SS outer -1.56"
104 cz 68.0174 $ 5 cm side reflector
105 cz 93.4974 $ 12 inch side reflector
c Axial - Bottom to Top of Cask
200 pz -211.201 $ 304 SST - 1.5"
201 pz -207.391 $ Cast DU - 3.75"
202 pz -197.866 $ 304 SST - 1.25"
203 pz 235.839 $ Cavity -169.5"
204 pz 238.379 $ 304 SST - 1"

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205 pz 245.999    $ Cast DU - 3"
206 pz 251.079    $ 304 SST - 2"
c
c Axial planes for spacer grid
c
450 pz 199.3265 $ Top of top spacer
451 pz 194.8815
452 pz 152.0063
453 pz 147.5613
454 pz 117.4115
455 pz 115.5065
456 pz 85.4837
457 pz 81.0387
458 pz 50.8889
459 pz 48.9839
460 pz 18.9611
461 pz 14.5161
462 pz -15.6337
463 pz -17.5387
464 pz -47.5615
465 pz -52.0065
466 pz -114.0841
467 pz -118.5291
468 pz -174.9552
469 pz -180.6702 $ 1st spacer from bottom
1000 so 1000

c Translations for horizontal row of 6 poison pins
tr1 -9.525 0.8763 0 $ Left
tr2 -5.715 0.8763 0 $
tr3 -1.905 0.8763 0 $
tr4 1.905 0.8763 0 $
tr5 5.715 0.8763 0 $
tr6 9.525 0.8763 0 $ Right
c Translations for vertical row of 6 poison pins
tr11 1.18745 -9.525 0 $ Bottom
tr12 1.18745 -5.715 0 $
tr13 1.18745 -1.905 0 $
tr14 1.18745 1.905 0 $
tr15 1.18745 5.715 0 $
tr16 1.18745 9.525 0 $ Top
c Translate horizontal poison pin rows
tr20 0.0000 11.3419 0 $ Horiz above FA#1
tr21 -13.6771 13.0151 0 $ Horiz below FA#2
tr22 13.6771 13.0151 0 $ Horiz below FA#3
tr23 -27.3542 11.3419 0 $ Horiz above FA#4
tr24 27.3542 11.3419 0 $ Horiz above FA#5
tr25 -27.3542 -13.0945 0 $ Horiz below FA#4
tr26 0.0000 -13.0945 0 $ Horiz below FA#1
tr27 27.3542 -13.0945 0 $ Horiz below FA#5
tr28 -13.6771 -14.7677 0 $ Horiz above FA#6
tr29 13.6771 -14.7677 0 $ Horiz above FA#7
c Translate vertical poison pin rows
tr30 -2.3352 26.1096 0 $ vert right of FA#2
tr31 -0.0397 26.1096 0 $ vert left of FA#3
tr32 -16.0123 0.0000 0 $ vert right of FA#4
tr33 -13.7168 0.0000 0 $ vert left of FA#1
tr34 11.3419 0.0000 0 $ vert right of FA#1
tr35 13.6374 0.0000 0 $ vert left of FA#5
tr36 -2.3352 -26.1096 0 $ vert right of FA#6
tr37 -0.0397 -26.1096 0 $ vert left of FA#7
c FA translations
tr121 -13.6771 26.1096 0 $ FA #2
tr122 13.6771 26.1096 0 $ FA #3
tr123 -27.3542 0 0 $ FA #4
tr124 27.3542 0 0 $ FA #5
tr125 -13.6771 -26.1096 0 $ FA #6
tr126 13.6771 -26.1096 0 $ FA #7
c Shift FA within basket channel

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trl32 -0.478 0.478 0 $ FA #2
trl33 0.478 0.478 0 $ FA #3
trl34 -0.478 0 0 $ FA #4
trl35 0.478 0 0 $ FA #5
trl36 -0.478 -0.478 0 $ FA #6
trl37 0.478 -0.478 0 $ FA #7
mode n
kcode 3000 1 30 240 50000
print -128
prtmp j -120 j 3
fc4 fission * lethargy**2 in cell 1
f4:n 1
fm4 -1 1 -6
sd4 1.0
c lethargy**2 [base 20 MeV -- u=0 at 20 MeV]
de4 1.e-8 1.e-7 1.e-6 1.e-5 1.e-4 .001 .01 .1 1 10 14
df4 458.7 365.3 282.6 210.5 149.0 98.08 57.77 28.07 8.974 0.48 .127
fc14 fission in cell 1
f14:n 1
fm14 -1 1 -6
sd14 1.0
fc24 fission * lethargy in cell 1
f24:n 1
fm24 -1 1 -6
sd24 1.0
c lethargy [base 20 MeV -- u=0 at 20 MeV]
de24 1.e-8 1.e-7 1.e-6 1.e-5 1.e-4 .001 .01 .1 1 10 14
df24 21.42 19.11 16.81 14.51 12.21 9.90 7.60 5.298 2.996 0.693 0.357
c m1 is UO2 fuel - Enriched to 4.25 wt% U235
m1 92235.50c -0.037462
92238.50c -0.843989
8016.50c -0.118549
c m2 is He
m2 2004.50c -1.0
c m3 is Zircaloy 4
m3 40000.60c -98.18
8016.50c -0.12
24000.50c -0.10
26000.55c -0.20
50000.35c -1.40
c m4 is water
m4 1001.50c 0.666700 8016.50c 0.333300
c m5 is Natural UO2 blanket fuel - 0.72 wt% U235
m5 92235.50c -0.0062672
92238.50c -0.875206
8016.50c -0.118528
c m6 is 304L SS
m6 6000.50c -0.03
14000.50c -1
15031.50c -0.045
16032.50c -0.03
24000.50c -20
25055.50c -2
26000.55c -64.895
28000.50c -12
c m7 is Inconel 718
m7 5000.01c -0.006
6000.50c -0.08
13027.50c -0.8
14000.50c -0.35
22000.50c -1.15
24000.50c -21
25055.50c -0.35
26000.55c -11.164
27059.50c -1
28000.50c -55
29000.50c -0.3
41093.50c -5.5
42000.50c -3.3

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c	m8 is Inconel X750	
m8	6000.50c	-0.08
	13027.50c	-1
	14000.50c	-0.5
	22000.50c	-2.75
	24000.50c	-15
	25055.50c	-1
	26000.55c	-7
	27059.50c	-1
	28000.50c	-70
	29000.50c	-0.5
	41093.50c	-1.2
c	m10 is B4C poison	
m10	5010.50c	-0.140886 \$ B10 lower end 18.3% - 0.3%
	5011.50c	-0.641812 \$ B11 upper end 81.7% + 0.3%
	6000.50c	-0.217302 \$ C
c	m11 is Depleted Uranium	
m11	92235.50c	0.000106128
	92238.50c	0.0475275
c	m12 is Air	
m12	7014.50c	-0.765
	8016.50c	-0.235
c	m13 is Channel - 304 SS and water mixture (VF water = 0.358)	
m13	1001.50c	-0.007266
	6000.50c	-0.000748
	8016.50c	-0.057669
	14000.50c	-0.009351
	15031.50c	-0.000421
	16032.50c	-0.000281
	24000.50c	-0.187013
	25055.50c	-0.018701
	26000.55c	-0.620369
	28000.50c	-0.098182
mt4	lwtr.01t	
mt13	lwtr.01t	