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Preliminary Neutronics Calculations for the Super SARA Test Program

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Pacific Northwest Laboratory
Operated by
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Commission

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ABSTRACT

The Super SARA Test Program (SSTP) will simulate small-break loss-of-coolant accident events in light-water reactors. Neutronics analyses have been performed in support of the SSTP in the ESSOR reactor. This report contains calculated power distributions for uniformly enriched UO_2 test assemblies and power coupling factors for the test assemblies relative to the driver assemblies.

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PRELIMINARY NEUTRONICS CALCULATIONS FOR THE SUPER SARA TEST PROGRAM

SUMMARY

Preliminary Neutronics analyses have been performed for loss-of-coolant accident experiments in the ESSOR reactor. The calculations were performed to determine the fuel enrichment for the test assembly and the power distribution across the test assembly. Power distributions were calculated for uniformly enriched UO_2 fuel assemblies of 2% and 3% ^{235}U . Edge-to-center power ratios are 2.0 and 2.3, respectively. Evaluations have not been carried out to determine whether these power ratios are acceptable.

The analyses indicate that 2% enriched fuel will generate sufficient power to obtain the fuel deformation desired for the experiment. For a reactor power of 22 MW, the average linear power for a 3% enriched test fuel assembly is 26.7 kW/m. For 2% test fuel, the average linear power is reduced by 21%. These results are based on fresh driver fuel and no control rods to compensate for excess reactivity. The results contained herein are preliminary and should be used as a base for completing the neutronic analyses. The results are not definitive enough to address all aspects of the experiment.

Additional analyses required for designing the experiment are a benchmark calculation for evaluation of the calculational methods, calculations with steam coolant to evaluate transient aspects of the experiment, graded enrichment test-assembly designs to flatten the power, elimination of excess reactivity from the reactor model, and axial calculations to predict and evaluate effects in the vertical direction.

INTRODUCTION

The Super SARA Test Program (SSTP)(1,2) is a major European Economic Community effort to study light-water reactor safety during large- and small-break loss-of-coolant accident (LOCA) events in the ESSOR reactor.(3) The SSTP will simulate small-break LOCA's by producing slow temperature ramps to high fuel rod temperatures (~2000°C). These experiments will provide data to describe high-temperature, severe-fuel damage mechanisms. As part of the U.S. Nuclear Regulatory Commission's (NRC) contribution to the SSTP, Pacific Northwest Laboratory (PNL) was contracted to supply analytical and engineering support including test-train design. In support of the test assembly design, a series of neutronics analyses were outlined. Because of a subsequent reduction in program funding, the neutronics analyses were terminated. This report provides the results of the calculations that had been made up to the point of termination.

The first section describes the reactor and test assembly with respect to neutronics analyses. The calculational methods are discussed in the second section. Results of the analyses are presented in the third section. Finally, the fourth section contains a brief discussion of calculations which should be done to complete the neutronics analyses if additional funding becomes available.

REACTOR AND TEST DESCRIPTION

REACTOR

The ESSOR reactor is a D₂O moderated experimental facility located in Italy. A horizontal section through the ESSOR reactor is shown in Figure 1. The 12 large circles in the center of the reactor are positions in which experiments can be performed. The SUPER SARA high-temperature, high-pressure loop is in Position 5. During the small-break LOCA experiments the other positions will contain D₂O although driver fuel could be placed in them if needed. Surrounding the test positions are 16 driver fuel assemblies. Outside the driver fuel assemblies are control rods, safety rods, and compensation rods. These were assumed to be withdrawn for calculational purposes.

Detailed position specifications are given in Figure 2. In particular, note that the SUPER SARA loop is 18.1 cm off center. The moderator is D₂O at a temperature of 325°K and a density of 1.1 g/cm³.

DRIVER ASSEMBLY

A cross-sectional view of a driver fuel assembly is shown in Figure 3. The fuel plates are of the MTR type with fully enriched uranium-aluminum alloy clad in aluminum. Twenty-four wt% of the alloy is uranium. Each assembly contains 513 grams of uranium. The active length is 154 cm. All other structural material is aluminum except for the 2-mm-thick outer tube which is Zircaloy-2. The D₂O coolant temperature is 325°K and the density is 1.1 g/cm³.

TEST ASSEMBLY

A cross-sectional view of the test assembly is shown in Figure 4. The 32 fuel rods represent a portion of a 17 x 17 assembly. The data are given in Table 1. Normal operating conditions are assumed. The active length is 200 cm, some 46 cm longer than the driver fuel. For the given pressure and temperature within the loop the light-water density is 0.7116 g/cm³. The octagonal region surrounding the test pins is a ZrO₂ shroud (density of 0.96 g/cm³) clad with Zirc-4. It is cooled with water at the same pressure and temperature as the fuel rods. The next annulus is the pressure vessel consisting of Zr-2½ Nb material. The outer annulus is a safety vessel. Again the material is Zr-2½ Nb. Between the pressure vessel and safety vessel is CO₂ coolant. Dimensions of the shroud, pressure vessel, and safety vessel are given in Figure 4.

- Test Position (5 → Super Sara Test Position)
- ⊙ Driver Fuel
- Control, Safety, Compensation Rods

FIGURE 1. Schematic Horizontal Section Through ESSOR Reactor Core

ELEMENT	ANGLE (degrees)	Element No.	Description
1	0°	1, 2, 3	Test Position
2	26° 33'	4, 5, 6, 7	Driver Fuel
3	63° 27'	8, 9, 10, 11	Control, Safety, and Compensation Rods
4	12° 10'		
5	77° 50'		
6	37° 40'		
7	52° 20'		
8	50° 50'		
9	39° 10'		
10	78° 40'		
11	11° 20'		

Radii units are millimeters

FIGURE 2. Geometry of ESSOR

(Dimensions in millimeters)

Plate No.	Radius, mm	
	Internal	External
1	15.976	17.246
2	20.243	21.513
3	24.511	25.781
4	28.778	30.048
5	33.045	34.315
6	38.080	39.350

FIGURE 3. Driver Fuel Assembly

FIGURE 4. ESSOR Small-Break LOCA Experiment Cross Section

TABLE 1. Test Fuel Data

Parameter	Value
Pellet OD, in. (mm)	0.3225 (8.19)
Pellet Density, %T.D.	95
Clad ID, in. (mm)	0.329 (8.36)
Clad OD, in. (mm)	0.374 (9.50)
Clad material	Zirc-4
Pitch, in. (mm)	0.496 (12.6)
Coolant Pressure, psia	2250
Coolant Temperature, °K	581
Fuel Temperature, °K	978

CALCULATIONAL METHODS

The neutronics methods used to model the experiments and the reactor are discussed in this section.

CODES

Several neutronics codes were used to determine reactivity, spatial neutron flux distributions, power distributions, and reaction rates. The NITAWL Code⁽⁴⁾ was used to self-shield resonances of the actinide isotopes and to produce AMPX⁽⁵⁾ working libraries. The XSDRNPM Code⁽⁶⁾ was used to perform one-dimensional, discrete ordinates calculations with anisotropic scattering. It produces a weighted library starting with either a working library or a weighted library. XSDRNPM was used for "unit cell," single assembly, and whole reactor calculations. An S_4 quadrature set was used with first-order anisotropy. The 2DBSS2 diffusion-theory code⁽⁷⁾ was used for one- and two-dimensional reactor calculations. Card-image cross-section data for input to 2DBSS2 was extracted from the weighted library produced by XSDRNPM. The DASQHE Code⁽⁸⁾ was used to determine the Dancoff correction for the test fuel rods.

CROSS SECTIONS

The cross sections⁽⁹⁾ used in the study were derived from ENDF/B-IV data⁽¹⁰⁾ via the AMPX system by Oak Ridge National Laboratory (ORNL). The 218-group data was collapsed to 27-group data. The materials and ID numbers used in this study are listed in Table 2. Data were not available for Zirc-4, so the Zirc-2 data were used because the cross sections are nearly the same.

TABLE 2. Materials Used in the Analyses

<u>Material</u>	<u>ID Number</u>
¹⁶ O	1276
H	1269
Zr	714102
²³⁵ U	1261
²³⁸ U	1262
D	112003
⁹³ Nb	1189
Zirc-2	128402
²⁷ Al	1193

GROUP STRUCTURES

The 27-group neutron structure is given in Table 3. It was used for "unit-cell" and single-assembly calculations. The four-group neutron structure is listed in Table 4. It was used for one-dimensional and two-dimensional reactor calculations.

DRIVER ASSEMBLY

The calculational sequence for the driver assembly is a NITAWL and XSDRNPM calculation for the fuel plate and associated coolant (D_2O) followed by an XSDRNPM calculation of the whole assembly. Since the 0.2 cm of Zircaloy-2 at the edge of the assembly is such a small fraction of the assembly and has a negligible effect neutronically, it was ignored in the assembly calculation. The XSDRNPM calculations were done with 27 energy groups. The results of the assembly calculation were collapsed to four groups and a set of cell-averaged macroscopic cross sections were generated for 2DBSS2 with WGT2DB.

Cross sections for the fuel plates and associated coolant were obtained assuming an infinite array of flat plates with 0.2898 cm of D_2O between them. A Dancoff correction factor of 0.818 was used in the resonance calculations for ^{235}U and ^{238}U . The flux-weighted microscopic cross sections were combined with the volume-weighted densities put out by XSDRNPM to make up the fuel region in the assembly calculation.

The assembly calculation consists of five concentric regions as listed in Table 5. Volumes of the various materials were conserved. The number of intervals used to describe each region is also given in Table 5. Region 3 is the fuel plate and associated coolant mixtures. A reflecting boundary was used at the outside edge.

TEST ASSEMBLY

A "unit-cell" calculation of the fuel rod and associated water (H_2O) was done with NITAWL and XSDRNPM in cylindrical geometry with reflecting boundaries. The outer radius was 0.7108 cm. To simplify the calculation, the gap and the clad were smeared together. The Dancoff correction was determined with DASQHE assuming an infinite array of rods. Flux-weighted 27-group microscopic cross sections were output by XSDRNPM.

The test assembly model consists of four concentric regions as listed in Table 6. The flux-weighted cross sections combined with the volume-weighted densities of the "unit-cell" calculation describes Region 1. Region 2 is a homogeneous region of the shroud and all light water not associated with the "unit cells." Region 3 is a homogeneous mixture of the pressure vessel, the safety vessel and the gap between them. The CO_2 in the gap has been ignored. Region 4 is D_2O moderator. Its radius is based on the center-to-center spacing of the test positions. The number of intervals used to describe each region is also given in Table 6. The 27-group XSDRNPM

TABLE 3. Twenty Seven-Group Neutron Structure

Group Number	Energy Range, eV	Lethargy Width
1	2.000000+007---6.434000+006	1.1341
2	6.434000+006---3.000000+006	0.7630
3	3.000000+006---1.850000+006	0.4834
4	1.850000+006---1.400000+006	0.2787
5	1.400000+006---9.000000+005	0.4418
6	9.000000+005---4.000000+005	0.8109
7	4.000000+005---1.000000+005	1.3863
8	1.000000+005---1.700000+004	1.7720
9	1.700000+004---3.000000+003	1.7346
10	3.000000+003---5.500000+002	1.6965
11	5.500000+002---1.000000+002	1.7047
12	1.000000+002---3.000000+001	1.2040
13	3.000000+001---1.000000+001	1.0986
14	1.000000+001---3.050000+000	1.1874
15	3.050000+000---1.770000+000	0.5442
16	1.770000+000---1.300000+000	0.3086
17	1.300000+000---1.130000+000	0.1402
18	1.130000+000---1.000000+000	0.1222
19	1.000000+000---8.000000-001	0.2231
20	8.000000-001---4.000000-001	0.6932
21	4.000000-001---3.250000-001	0.2076
22	3.250000-001---2.250000-001	0.3677
23	2.250000-001---1.000000-001	0.8109
24	1.000000-001---5.000000-002	0.6931
25	5.000000-002---3.000000-002	0.5108
26	3.000000-002---1.000000-002	1.0986
27	1.000000-002---1.000000-005	6.9077

TABLE 4. Four-Group Neutron Structure

<u>Group Number</u>	<u>Energy Range, eV</u>	<u>Lethargy Width</u>
1	2.00+7---1.00+5	5.2983
2	1.00+5---5.50+2	5.2030
3	5.50+2---3.05+0	5.1948
4	3.05+0---1.00-5	12.6280

TABLE 5. One-Dimensional Model of Driver Fuel Assembly

<u>Region Number</u>	<u>Material</u>	<u>Radius, cm</u>	<u>Number of Intervals</u>
1	D ₂ O	1.2487	4
2	Al	1.9292	2
3	Fuel	4.0484	9
4	Al	4.4351	1
5	D ₂ O	11.6	14

TABLE 6. One-Dimensional Model of Test Assembly

<u>Region Number</u>	<u>Material</u>	<u>Radius, cm</u>	<u>Number of Intervals</u>
1	Fuel	4.0208	8
2	Shroud	5.64	3
3	Vessels	8.26	5
4	Moderator	14.44	14

results were collapsed to four groups on a region-wise basis; i.e., the spatial weighting takes place only within the region. Using region-weighting allows one to represent each of the regions in succeeding calculations. The four-group microscopic cross sections were converted to 2DBSS2 format using WGT2DB.

REACTOR

Both one- and two-dimensional models of the ESSOR reactor were generated. To model the experiment in one dimension, the center of the test assembly was chosen as the origin; i.e., the test assembly was displaced to the center of the reactor. The first four regions are the same as that used

in the test-assembly model described earlier. Region 5 contains D₂O moderator. Region 6 is the homogenized driver fuel. The inside radius is 49.04 cm and the outside radius is 67.51 cm. Region 7 contains D₂O and has an outside radius of 90 cm. One-dimensional four-group calculations were performed with 2DBSS2 as well as XSDRNP. An axial buckling of $6.9 \times 10^{-5} \text{ cm}^{-2}$ was incorporated in the 2DBSS2 calculation in order to represent axial leakage.

The reactor was modeled in X-Y geometry as indicated in Figure 5 with half-core symmetry. Each tick on the ordinate and abscissa represents a mesh interval. Dimensions are given along the edge of the diagram. Interval thickness is uniform between any given pair of dimensions. The test fuel is represented (Region 1) as a homogeneous mixture of fuel, clad, and coolant. Each rod is represented by 2 x 2 mesh intervals. The shroud and vessels (Regions 2 and 3, respectively) were modeled in a rectangular fashion with a constant thickness. The respective volumes are the same as in the 1-D annular model. Regions 5 through 10 represent the driver fuel, with Regions 6 and 10 representing two driver assemblies each and Regions 5, 7, 8, and 9 representing one driver assembly each. Again, volume is conserved. The driver assemblies are located in the approximate positions as indicated in Figure 2. Region 4 is D₂O moderator. The axial buckling was assumed to be $6.9 \times 10^{-5} \text{ cm}^{-2}$.

FIGURE 5. ESSOR X-Y Model

RESULTS

Results of the neutronics analyses are presented in this section.

FLUX DISTRIBUTIONS

The effect of the test assembly on the thermal flux distribution is shown in Figure 6. The comparison is based on XSDRNPM calculations normalized to one source neutron. The test-fuel enrichment is 3%. Note that the effect of the test fuel is negligible near the driver fuel.

A check on the accuracy of diffusion theory was made by comparing the thermal flux shape from the one-dimensional 2DBSS2 calculation with the thermal flux shape from the XSDRNPM calculation (Figure 7). Both calculations were normalized to one source neutron. The codes agree on the general shape but there are obviously some detailed discrepancies. The thermal flux in the test fuel is slightly higher for the XSDRNPM calculation. Further comparisons will be presented in later subsections.

POWER DISTRIBUTION

The power distribution in a fuel rod is shown in Figure 8 for 3% enrichment. The results are from the "unit cell" calculation, so the effect of the surrounding D₂O is not taken into account. For the fuel rods on the edge of the test assembly, the neutron spectrum will be more thermalized and the surface-to-center power ratio will be higher. Based on the unit cell calculation the surface-to-center power ratio is 1.12 for 3% enrichment; for 2% enrichment the ratio is 1.09.

One-dimensional radial power profiles in the test assembly are tabulated in Table 7. The 27-group results are from the test assembly calculation, whereas the four-group results are from the reactor calculation. Since the two-dimensional calculation entails diffusion theory with four energy groups, the change in power distribution from the second to the third to the fourth column of Table 7 gives an indication of the accuracy to be expected from the two-dimensional calculation. This presumes that the 27-group transport calculation is exact. The effect of 4 groups versus 27 groups is 2.7% at the most (last interval). A graphical comparison is shown in Figure 9. The plotted data extrapolated to the edge of the fuel gives a discrepancy of 8.5%. It is probably more realistic to use the average of the interval because the fuel is modeled as a homogeneous mixture of fuel, clad, and coolant. The power in the last interval represents nearly 8 fuel rods out of 32; i.e., 25% of the volume. Using four energy groups seems to be adequate.

The effect of using diffusion theory rather than transport theory is shown in Figure 10. Results from 2DBSS2 are in good agreement with the results from XSDRNPM. The largest discrepancy (5%) is at the center of the test assembly. It represents a very small fraction of the total power. The

FIGURE 6. ESSOR Thermal Flux Distribution

FIGURE 7. Thermal Flux Shape in ESSOR with Test Assembly

FIGURE 8. Power Distribution in a Test Fuel Rod

TABLE 7. Radial Power Profile in Test Assembly

Radius, cm	3% Enriched			2% Enriched	
	27 Gp	4 Gp	4 Gp	27 Gp	4 Gp
0.251	0.6059	0.6054	0.5775	0.6732	0.6593
0.754	0.6240	0.6256	0.5981	0.6896	0.6778
1.256	0.6608	0.6664	0.6411	0.7224	0.7147
1.759	0.7195	0.7302	0.7097	0.7739	0.7718
2.262	0.8062	0.8221	0.8097	0.8481	0.8526
2.764	0.9324	0.9492	0.9489	0.9526	0.9620
3.267	1.1130	1.1224	1.1388	1.0971	1.1073
3.770	1.4133	1.3763	1.3950	1.3237	1.3121
CODE	XSDRNPM	XSDRNPM	2DBSS2	XSDRNPM	XSDRNPM

FIGURE 9. Effect of Group Structure on Power Profile for 32-Pin Test Assembly With Water Coolant

FIGURE 10. Effect of Code on Power Profile for 32-Pin Test Assembly with Water Coolant

small discrepancy toward the edge of the assembly is in the opposite direction of the discrepancy in going from 27 to 4 energy groups.

The ratio of power in the last interval to the first interval is 2.27 for 3% enriched test fuel (four energy groups, XSDRNPM). For 2% enriched test fuel the ratio is 1.99. Thus, lowering the enrichment reduces the power peaking. The effect of changing enrichment is shown in Figure 11.

The power distribution calculated in two dimensions is shown in Figure 12 for 3% enriched test fuel. The rod powers at the bottom of the figure are closest to the reactor center. The ratio of maximum-to-minimum rod power is 2.17, nearly the same as the one-dimensional results. The power tilt across the assembly is negligible. The power tilt across the core is only about 1%.

POWER LEVEL

The absolute power level in the test assembly is very important during the preconditioning phase in order to obtain the desired fuel relocation. Since the axial direction was not represented in any of the calculational models, only the ratio of test assembly power-to-driver power was calculated. The values given in succeeding paragraphs were obtained by normalizing the driver power to 22 MW and taking an axial average; that is, averaging over a 154 cm length. The calculated ratios are given in Column 3 of Table 8.

The first three lines of Table 9 give results for 3% enriched test fuel using the various models. The average linear power is calculated to be 9.7 kW/ft using XSDRNPM results. Using the one-dimensional 2DBSS2 model lowers the linear power by 3% to 9.4 kW/ft. With the two-dimensional model the linear power is only 8.1 kW/ft or 13% less than the one-dimensional model. In any case, there is sufficient power in the test assembly to obtain the desired fuel preconditioning temperatures.

Results for 2% enriched test fuel are given on Line 4 of Table 8. If the test fuel enrichment is reduced to 2%, the linear power is reduced by 20%.

All of the previous results were based on calculations using fresh driver fuel. One additional calculation was done to determine the sensitivity of the test-fuel linear power to burnup of the driver fuel. Burnup was simulated by reducing the ^{235}U content of the driver fuel by 30%. The same four-group driver-fuel microscopic cross sections were used for simulated burned fuel as were used for fresh fuel. The average linear power in the test fuel increased by 43% (Line 5 of Table 8). This result demonstrates the necessity of describing the driver assemblies accurately.

REACTIVITY

The driver fuel contains very few absorbers other than ^{235}U . With reflecting boundaries; each assembly has a xenon-free neutron multiplication of 1.8727. As the assembly is irradiated the reactivity will decrease.

FIGURE 11. Effect of Enrichment on Power Profile for 32-Pin Test Assembly with Water Coolant

1.153	1.319	
0.758	0.921	1.320
0.608	0.760	1.156
0.609	0.760	1.156
0.761	0.923	1.321
1.155	1.320	

FIGURE 12. Relative Power Distribution for 3% Enriched Fuel in 32-Pin Test Assembly with Water Coolant

TABLE 8. Test Assembly Powers for 22 MW Reactor Power

Code	Enrichment %	Power Ratio	Average Linear Power	
			w/cm	kW/ft
XSDRNPM	3	0.0712	318	9.7
2DBSS2 (1-D)	3	0.0687	307	9.4
2DBSS2 (2-D)	3	0.0564	267	8.1
XSDRNPM	2	0.0562	251	7.6
XSDRNPM(a)	3	0.1015	453	13.8

(a) Driver fuel enrichment reduced by 30%.

TABLE 9. Reactivity Values of Test Fuel

Enrichment, %	Rod	Assembly
2	1.2040	1.0464
3	1.2943	1.1713

In generating cross sections for the test assembly, k_{∞} values were obtained for the fuel rod and associated water, and for the whole test assembly. The values are given in Table 9 for each enrichment. Since k_{∞} is greater than 1.0 in both cases, the assembly does not have to be driven. This could be the reason for the high test-assembly power. For steam coolant instead of water coolant, the rod and assembly k_{∞} values are expected to be much lower. However, because of interactions with the rest of the reactor, the void coefficient is expected to be positive. H_2O has a much larger thermal absorption cross section than D_2O and there is a sufficient amount of D_2O to thermalize the neutrons, so the loss of H_2O results in an increase in reactivity.

Reactor k_{eff} values are listed in Table 10. The one-dimensional 2DBSS2 value is somewhat lower than the XSDRNPM value; however, the two-dimensional 2DBSS2 value is much higher. The apparent discrepancy has not been investigated in detail. The effect of the test fuel enrichment on reactor k_{eff} is minimal as seen by comparing Lines 1 and 4 of Table 10. If all fresh driver fuel is used in ESSOR, a great deal of reactivity will have to be compensated for by poison. Line 5 of Table 10 shows the effect of simulated burned driver fuel. A 30% reduction in ^{235}U content reduces k_{eff} by 13%. If fission products are taken into account there will be an estimated 5% further reduction in k_{eff} .

REACTION RATES

Occasionally it is instructive to examine various reactions as functions of space and energy. Some of this data is presented in this subsection. All of the results pertain to 3% enriched test fuel.

Spatial effects within the test-fuel unit cell and assembly are given in Table 11. The table shows that 95% of the absorptions in the unit cell take place in the fuel. The total flux is flat across the unit cell but the thermal flux shows a depression in the fuel. Group 24 is representative of the thermal flux. The fuel shows a 10% depression relative to the water. At lower energies the effect would be larger. The effects within a region would be superimposed on the results in the top part of Table 11.

As shown in Table 11, most of the absorptions in the assembly take place in the fuel. However, a significant fraction also takes place in the pressure and safety vessels. Region-average spatial flux effects are also given in Table 11. The Group 24 flux is similar to the flux shape shown in Figure 6.

TABLE 10. Reactor Neutron Multiplication

Code	Test Fuel Enrichment, %	keff
XSDRNPM	3	1.2155
2DBSS2 (1-D)	3	1.1829
2DBSS2 (2-D)	3	1.3174
XSDRNPM	2	1.2065
XSDRNPM ^(a)	3	1.0872

(a) Driver fuel enrichment reduced by 30%.

TABLE 11. Spatial Effects in the Test Assembly
(3% enriched; water coolant; XSDRNPM results)

Region	Absorption	Total Flux	Group 24 Flux	Volume Fraction
Unit Cell				
Fuel	0.9516	0.9983	0.9346	0.3321
Clad	0.0103	0.9993	0.9935	0.1145
Water	0.0391	1.0011	1.0406	0.5534
Assembly				
Homogeneous Fuel	0.7448	0.7992	0.4406	0.0775
Water/Shroud	0.0684	0.9513	0.8024	0.0750
Vessels	0.1835	0.9956	0.9417	0.1747
Moderator	0.0057	1.0297	1.1016	0.6728

The effect of the shroud, vessels, and D₂O moderator on the flux spectrum is shown in Table 12 by comparing group fluxes from the unit cell calculation to group fluxes from the assembly calculation. Each set of numbers was normalized to 1.0. At high energies the assembly flux is 30% lower than the unit cell flux. As thermal energies are approached, the assembly flux becomes higher than the unit cell flux. The fission rate as a function of energy shows similar characteristics; however, since the fission cross section is largest at thermal energies, the normalization gives a factor of three fewer fissions in the assembly at high energies.

TABLE 12. Energy Dependency of Flux and Fission Rate
(3% enriched, water coolant, XSDRNPM results)

Group Number	Flux		Fission Rate	
	Unit Cell	Region 1 of Assembly	Unit Cell	Assembly
1	0.0072	0.0055	0.0046	0.0013
2	0.0542	0.0477	0.0211	0.0069
3	0.0685	0.0587	0.0258	0.0083
4	0.0422	0.0363	0.0111	0.0036
5	0.0632	0.0542	0.0033	0.0010
6	0.1187	0.0883	0.0029	0.0008
7	0.1157	0.0810	0.0031	0.0008
8	0.0852	0.0573	0.0033	0.0008
9	0.0660	0.0442	0.0046	0.0012
10	0.0604	0.0410	0.0098	0.0025
11	0.0556	0.0387	0.0202	0.0053
12	0.0349	0.0254	0.0258	0.0070
13	0.0298	0.0227	0.0241	0.0069
14	0.0279	0.0226	0.0181	0.0055
15	0.0141	0.0117	0.0039	0.0012
16	0.0081	0.0068	0.0026	0.0008
17	0.0036	0.0031	0.0041	0.0013
18	0.0031	0.0027	0.0055	0.0018
19	0.0059	0.0051	0.0067	0.0022
20	0.0191	0.0169	0.0284	0.0094
21	0.0056	0.0052	0.0154	0.0053
22	0.0102	0.0111	0.0363	0.0149
23	0.0291	0.0597	0.1139	0.0872
24	0.0329	0.1063	0.1971	0.2376
25	0.0194	0.0717	0.1589	0.2196
26	0.0161	0.0636	0.1878	0.2763
27	0.0032	0.0124	0.0615	0.0906

The spatial effect of the flux as a function of energy is shown in Table 13 for the test assembly. At high energies the flux is highest in the center of the assembly. As energy decreases, the peak flux shifts to the edge of the assembly. At thermal energies, the flux ratio approaches three.

Four-group reaction rates are listed in Table 14 for the ESSOR reactor with the 3% enriched test assembly. The data in Table 14 is normalized to one source neutron. The test assembly is broken down into the homogenized fuel and the shroud and vessels. The flux values given are actually flux times volume. For every 0.06666 neutrons born in the test assembly there are 0.05018 absorbed in the test fuel and 0.01767 absorbed in the shroud and vessels, or a total of 0.06785 absorbed in the assembly. The sum of the leakage terms is the net number of neutrons leaving the test assembly. The negative value of -0.00112 says there are more neutrons entering the test assembly than leaving the test assembly. It amounts to less than 2% of the absorptions. The group leakage values illustrate that most of the fast neutrons leave the test fuel to be thermalized in the D₂O moderator. A large fraction then re-enter the test fuel as thermal neutrons. Note that there are only 1.996 source neutrons for each fission in the test fuel whereas ν for ²³⁵U is 2.42. The ratio is the k_{eff} value of the reactor.

The bottom part of Table 14 is a summary for the reactor, including the test assembly. The sum of absorptions plus leakage should equal the source. The discrepancy is attributed to numerical inaccuracies. Leakage from the reactor accounts for 32% of the neutrons. Most of this occurs in the thermal group. The leakage is high because of the thin annulus of driver fuel. The actual geometry of cylindrical elements may lend itself to still more leakage. Most of the absorptions and fissions also take place in the thermal group. The thermal flux is much higher than the nonthermal flux because of the large amount of D₂O.

Energy Group	Test Fuel	Shroud & Vessels	Leakage	Source
Fast	0.05018	0.01767	-0.00112	1.00000
Thermal	0.01767	0.00000	0.00000	0.01767
Total	0.06785	0.01767	-0.00112	1.01767
Fast	0.05018	0.01767	-0.00112	1.00000
Thermal	0.01767	0.00000	0.00000	0.01767
Total	0.06785	0.01767	-0.00112	1.01767
Fast	0.05018	0.01767	-0.00112	1.00000
Thermal	0.01767	0.00000	0.00000	0.01767
Total	0.06785	0.01767	-0.00112	1.01767
Fast	0.05018	0.01767	-0.00112	1.00000
Thermal	0.01767	0.00000	0.00000	0.01767
Total	0.06785	0.01767	-0.00112	1.01767
Fast	0.05018	0.01767	-0.00112	1.00000
Thermal	0.01767	0.00000	0.00000	0.01767
Total	0.06785	0.01767	-0.00112	1.01767

TABLE 13. Energy/Space Dependency of Flux in the Test Assembly
(3% enriched, water Coolant, XSDRNPM results)

Group Number	Assembly Flux		Average
	Space Point 1	Space Point 8	
1	0.174	0.143	0.164
2	1.514	1.237	1.425
3	1.867	1.530	1.755
4	1.160	0.948	1.086
5	1.747	1.409	1.620
6	2.881	2.322	2.640
7	2.619	2.204	2.422
8	1.807	1.633	1.712
9	1.356	1.299	1.320
10	1.232	1.230	1.227
11	1.141	1.177	1.157
12	0.727	0.794	0.758
13	0.639	0.718	0.679
14	0.624	0.727	0.676
15	0.3273	0.3663	0.3492
16	0.1916	0.2124	0.2034
17	0.0867	0.0967	0.0922
18	0.0752	0.0844	0.0802
19	0.1422	0.1582	0.1511
20	0.4741	0.5333	0.5063
21	0.1441	0.1681	0.1567
22	0.2874	0.3753	0.3331
23	1.267	2.248	1.784
24	1.925	4.366	3.179
25	1.197	3.107	2.144
26	0.993	2.915	1.900
27	0.1887	0.6026	0.3720

TABLE 14. Reaction Rates in ESSOR Experiment (3% enriched, water coolant, XSORNPM results)

	Group Number	Source	Absorb	Leakage	Fission	Flux
Test Fuel (Region 1)	1	0.06566	0.00108	0.04950	0.00074	0.3782
	2	0.00099	0.00089	0.00419	0.00012	0.1189
	3	0	0.00284	0.00077	0.00053	0.0740
	4	0	0.04538	-0.03789	0.03201	0.3997
	Total	0.06666	0.05018	0.01657	0.03339	0.9708
Shroud and Vessels (Regions 2 and 3)	1	0	0.00011	-0.00811	0	0.5754
	2	0	0.00070	0	0	0.3085
	3	0	0.00066	0.00134	0	0.2128
	4	0	0.01620	-0.01092	0	2.8647
	Total	0	0.01767	-0.01769	0	3.9614
Reactor	1	0.98507	0.00560	0.00243	0.00153	20.06
	2	0.01493	0.00413	0.01329	0.00173	30.10
	3	0	0.01827	0.02238	0.00870	29.40
	4	0	0.65557	0.28189	0.49046	362.54
	Total	1.00000	0.68357	0.31999	0.50243	442.10

TABLE 14. Reaction Rates in ESSOR Experiment (3X enriched, water coolant, 4200W/M results)

		Group Number	Source	Absorb	Leakage	Fission	Flux
Test Fuel (Region 1)		1	0.0656	0.0018	0.0450	0.00074	0.3735
		2	0.00099	0.00089	0.00419	0.00012	0.1789
		3	0	0.00294	0.0007	0.00023	0.0740
		4	0	0.04238	-0.03789	0.0201	0.3997
		Total	0.0666	0.0508	0.0457	0.03339	0.9708
Shroud and Vessels (Regions 2 and 3)		1	0	0.0001	-0.0081	0	0.2784
		2	0	0.00070	0	0	0.3085
		3	0	0.0066	0.00134	0	0.2128
		4	0	0.0150	-0.0105	0	2.8647
		Total	0	0.0157	-0.0169	0	3.9614
Reactor		1	0.98507	0.00560	0.00243	0.00123	20.06
		2	0.01493	0.00413	0.01329	0.00173	30.10
		3	0	0.01827	0.02238	0.00870	29.40
		4	0	0.62557	0.28189	0.49046	382.54
		Total	1.00000	0.68357	0.31999	0.50243	442.10

FUTURE ANALYSES

The results presented in the previous section were the initial efforts in designing the experiment. Additional analyses would be needed to go ahead with the experiment.

CORRELATION WITH NEUTRONICS EXPERIMENT

As the measured data becomes available from the neutronics experiment, analyses should be performed to benchmark the codes relative to the SSTP. These correlations would lend credibility to the physics models used for the SSTP. In particular, the calculated power coupling factor; i.e., the ratio of test-assembly power to driver-fuel power is subject to question because of the distance between the test fuel and the driver fuel.

STEAM COOLANT

During the transient phase of the experiment the test assembly will be cooled by steam. Calculations need to be done with steam coolant to determine the power coupling factor and the power distribution across the test assembly.

GRADED ENRICHMENT

The question of what power distribution is desired needs to be answered by iterating with thermal hydraulics because the temperature distribution is strongly related to the power distribution. If a flatter distribution is required, the spectrum entering the test assembly has to be hardened or the enrichment in the center of the test assembly has to be increased. In either case, further calculations would be needed.

LOWER REACTIVITY

As the experiment was modeled, there was quite a bit of excess reactivity. This excess reactivity has to be compensated for by using control rods or by using burned fuel elements. The amount of control needed has to be evaluated. In addition, the power coupling factor could be reduced because of fewer neutrons available for fission multiplication.

AXIAL CALCULATIONS

Thus far, no axial calculations have been done. Since the active length of the test fuel is longer than the active length of the driver fuel, some abnormal effects are expected near the end of the test fuel. If control rods are used to compensate for excess reactivity, axial calculations will have to

be done to evaluate the effects of partial insertion. If burned driver fuel is used, the ends of the elements will be more reactive than the centers. The effect on the experiment will probably be small but it should be evaluated. Calculations also should be done for cases where part of the test assembly is water cooled and part of the test assembly is steam cooled. This represents the reflood phase of the experiment.

CORRELATION WITH NEUTRONICS EXPERIMENT

As the measured data become available from the neutronics experiment, analyses should be performed to benchmark the codes relative to the SSTR. These correlations would lend credibility to the physics models used for the SSTR. In particular, the calculated power coupling factor, i.e., the ratio of test-assembly power to driver-fuel power, is subject to question because of the distance between the test fuel and the driver fuel.

STEAM COOLANT

During the transient phase of the experiment the test assembly will be cooled by steam. Calculations need to be done with steam coolant to determine the power coupling factor and the power distribution across the test assembly.

GRADED ENRICHMENT

The question of what power distribution is desired needs to be answered by iterating with thermal hydraulics because the temperature distribution is strongly related to the power distribution. If a flatter distribution is required, the spectrum entering the test assembly has to be hardened or the enrichment in the center of the test assembly has to be increased. In either case, further calculations would be needed.

LOWER REACTIVITY

As the experiment was modeled, there was quite a bit of excess reactivity. This excess reactivity has to be compensated for by using control rods or by using burned fuel elements. The amount of control needed has to be evaluated. In addition, the power coupling factor could be reduced because of lower neutrons available for fission multiplication.

AXIAL CALCULATIONS

Thus far, no axial calculations have been done. Since the active length of the test fuel is longer than the active length of the driver fuel, some abnormal effects are expected near the end of the test fuel. If control rods are used to compensate for excess reactivity, axial calculations will have to

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