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**CRITICALITY SAFETY ISSUES ASSOCIATED WITH
THE BURIAL OF HIGHLY ENRICHED NUCLEAR FUEL
IN A GEOLOGIC REPOSITORY**

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Managed by the U.S. Department of Energy

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**Westinghouse Idaho
Nuclear Company, Inc.**

PREPARED FOR THE
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1.0 Introduction

The purpose of this study is to provide some basic guidance regarding the criticality safety implications associated with long term burial of highly enriched nuclear fuel in a geologic repository. Generic calculations to determine the distance required to neutronically isolate neighboring fissile regions are reported. Two specific fuels are also addressed, Fort Saint Vrain and Shippingport PWR Core-2 Seed-2 (PWR).[†] These fuels are addressed under both dry and flooded conditions. Fort Saint Vrain fuel contains a uranium-graphite fuel matrix, is graphite moderated and has a relatively low ²³⁵U loading (~1 kg per element). PWR contains a uranium-zirconium fuel matrix, is water moderated, zircaloy clad and has a ²³⁵U loading of ~15 kg per cluster. These two fuels are representative of many of the fuels currently stored at the Idaho Chemical Processing Plant (ICPP).

2.0 Description

The spent fuel storage system under consideration¹ involves filling cylindrical canisters with spent nuclear fuel. These canisters are then to be placed into a geologic repository. Salt and tuff are the two most likely materials in which the canisters will be buried. Canisters will require sufficient material between them to ensure that the overall reactivity stays within acceptable limits. Fuels will also have to be shown to be critically safe under a variety of conditions. These conditions include flooding and the loss of the physical integrity of the fuel.

3.0 Requirements Documentation

Regulatory requirements directly related to criticality safety in a geologic repository are found in 10CFR60.131(b)(7)², DOE Order 5480.5³, and DOE Order 5480.24⁴. 10CFR60.131(b)(7) states:

“All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality is not possible unless at least two, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.”

Section 5.0, Discussion of Contingencies, of this report touches on some of the difficulties inherent in applying the double contingency principle to processes such as long term geologic repository storage.

4.0 Methodology

4.1 Codes

Three different criticality codes were used for this study, TWODANT, SCALE, and MCNP. TWODANT and SCALE were used to determine the amount of salt or tuff necessary to provide

[†]. Throughout this document the term “PWR” refers only to the Shippingport PWR Core-2 Seed-2 clusters.

isolation between neighboring fissile units. SCALE cross section preprocessing (NITAWL and BONAMI) was used to generate cross sections for the TWODANT calculations. The Hansen-Roach and ENDF/B-IV 27 group[†] cross sections were used for all TWODANT calculations.

TWODANT is a one or two dimensional, multigroup, discrete ordinates transport code. Revision 1.6 dated February 1, 1990 was used for this study. TWODANT is not statistical in nature and therefore better suited for calculations intended to define interaction between units than SCALE(with KENO) or MCNP.

The CSAS25 module of SCALE-4 (RCIC # CCC-545) was used for PWR calculations. The CSAS25 module consists of cross section preprocessing (NITAWL and BONAMI) followed by KENO-Va. KENO-Va is a monte-carlo based neutron transport code.

MCNP-4.2 was also used to analyze the Fort Saint Vrain fuel elements. MCNP is also a monte-carlo based transport computer code. MCNP offers more powerful geometry description capabilities than either SCALE or TWODANT. MCNP is capable of using ENDF/B-V cross section data that is continuous in energy.

4.2 Validation

References 5-12 demonstrate the applicability of MCNP and SCALE for highly enriched uranium systems. No validation for TWODANT exists at this time for the type of calculations performed in this study. Reference 13 does demonstrate the applicability of TWODANT for a highly enriched uranyl nitrate annular tank configuration. It should be pointed out that no experimental data exists for systems containing large amounts salt or tuff. In particular benchmark experiments involving large amounts of chlorine have not been performed and therefore those results involving chlorine must be taken with a grain of salt.

5.0 Discussion of Contingencies

This report constitutes a scoping study and therefore requires no contingency analysis. Safety analysis will require the application of the double contingency principle. Contingency analysis for permanent storage of fissile material in a geologic repository will be very different from how it is normally applied. Contingency failure will have to be shown to be unlikely for extended periods of time as opposed to relatively short periods of time. For example, if flooding is to qualify as an unlikely event it must be shown to have a probability of occurring below some predetermined value (typically 10^{-2} per year). Furthermore it will have to be shown that flooding remains unlikely throughout the lifetime of the repository.

Performing contingency analysis presumes some understanding of what constitutes normal/likely conditions. Events such as fuel and canister degradation, earthquake, flooding and human intrusion as well as other events would all require analysis to determine whether these events constitute a contingency failure or a normal and expected condition of operation.

In addition, a contingency failure normally results in some type of remedial action. This may not be feasible in a geologic repository. Recovery of a doubly contingent environment in the event of a contingency failure must be addressed.

[†]. This study led to the determination that the SCALE 27 group cross sections are inadequate for systems with large amounts of chlorine.

6.0 Evaluation and Results

6.1 Isolation of Fissile Regions

Calculations were performed to determine the amount of either salt or tuff required to isolate neighboring fissile regions. Infinite slabs of fissile material were assumed to be separated by infinite slabs of either salt or tuff. The fissile regions were assumed to be unmoderated ^{235}U . The resulting very fast neutron energy spectrum would serve to maximize the interaction between neighboring slabs. This approach is conservative and envelopes the cylindrical repository configuration. Salt and tuff compositions are found in appendix A. TWODANT was the code chosen used to determine reactivity. TWODANT gives results free of statistical uncertainty and is therefore well suited for clearly defining the interaction between fissile regions.

Table 1 shows the results of these calculations. It should be noted that the reactivity of these systems depends only on the areal densities of the materials. In particular, as long as the product of the material density and the thickness of that material is preserved, the reactivity will not change.

Table 1. Isolation of Fissile Regions with both Salt and Tuff. The areal density of the ^{235}U was assumed to be 3.3×10^{-2} atoms/b and 1.44×10^{-2} atoms/b for the salt and tuff cases respectively (the unit barn, abbreviated as b, is equal to 10^{-24} cm²).

thickness of medium between fissile regions (cm)	k_{eff} with salt (2.2g/cc)	k_{eff} with tuff (2.2g/cc)
100	1.1801	1.1397
125	1.0896	1.0479
150	1.0305	1.0058
175	1.0053	0.98887
200	0.99370	0.98274
225	0.98860	0.98070
250	0.98644	0.98006
275	0.98549	0.97987
300	0.98519	0.97982
∞	0.98497	0.97980

The results in Table 1 are shown pictorially in Figure 1. Figure 1 shows how the reactivity changes with medium thickness with respect to the reactivity when the medium is infinitely thick.

These results are conservative in that they incorporate two very significant assumptions;

- ☛ the fissile regions contain no moderating material
- ☛ the geometry is that of infinite alternating slabs

Any medium thickness derived from these results is therefore conservative for nearly all systems.

Figure 1. Interaction of Fissile Regions as a Function of the Thickness of the Medium between Fissile Regions.

6.2 Fort Saint Vrain

A cross section of a typical Fort Saint Vrain fuel element is shown in Figure 2. A typical fuel element¹⁴⁻¹⁶ is a hexagonally shaped piece of graphite with 325 holes drilled in it. Fuel rods are stacked in 210 of the holes while most of the rest of the holes are left open to allow the flow of coolant. Fuel holes are filled with a stack of ~2" long fuel rods. These fuel rods are cylindrical in shape and made up of coated fuel particles in a low density graphite matrix. At beginning of life ²³⁵U was the only fissile isotope present. A large thorium loading (~13 kg per element) resulted in the breeding of significant amounts of ²³³U. Maximum end of life loadings for one fuel element are 0.81 kg of ²³⁵U and 0.27 kg ²³³U.¹⁴ These loadings were used for all Fort Saint Vrain fuel elements modelled in this study. The typical Fort Saint Vrain fuel element also had two holes filled with boron in a graphite matrix to act as a neutron poison. These holes containing poison are located in two corners of the fuel element as shown in Figure 2.

Assumptions in the Fort Saint Vrain fuel element computer model include:

- ☛ Fuel regions were modelled as a homogeneous mixture of carbon, ²³³U, ²³⁵U, thorium, and silicon (silicon was used in the fuel particle coating).
- ☛ Carbon was distributed evenly everywhere except coolant/void channels.
- ☛ Boron was not included in any of the modelled elements.

The first set of Fort Saint Vrain calculations performed are shown pictorially in Figure 3. Calculated k_{eff} and the standard deviation are reported for various configurations. The figure also reports the minimum cylinder radius required to encompass the shown configuration. All of these calculations assume full water reflection and fuel elements stacked to an infinite height. The full water reflection assumption conservatively envelopes any increase in reactivity due to the storage canister and burial medium. The infinite height assumption, although conservative, is not overly so. Once the ratio of height to width exceeds ~3 the reactivity is nearly equivalent to that of a system that is infinite in length. Coolant channels within each element as well as other voids were either dry or filled with water (flooded).

Even with the full water reflection and infinite height assumptions the reactivity ($k_{eff}+2\sigma$) of seven dry elements (stacked to infinite height) is still only ~0.77. Allowing this system to become flooded with water, however, drives the reactivity well in excess of prompt critical. Flooding with pure water increased k_{eff} by 30 to 40%.

Flooding with relatively pure water, although likely in a tuff based burial site, is not likely in a salt environment. Flooding with a brine solution can have a very different effect on reactivity than pure water. The chlorine found in a brine solution, with its fairly large neutron absorption cross section, can significantly reduce the reactivity of a flooded system. As Table 2 and Figure 4 illustrate, as the chlorine density within the flooding water increases the reactivity of the system decreases. A nominal chlorine atom density of 3×10^{-3} atoms/b-cm (~150ppt) drives the reactivity of the 7 element flooded case to ~0.69, well below the ~0.77 obtained from the dry case.

- - void/coolant channel (mostly 0.625" diameter)
- - fuel (0.5" diameter) ⊞ - poison (0.5" diameter)
- - graphite

Figure 2. Cross Section of Typical Fort Saint Vrain Fuel Element. Total distance across flats is 14.172" (36.00 cm). Total height is 31.22" (79.30 cm). Typical void volume fraction is ~20%.

Figure 3. Reactivity of Fort Saint Vrain Elements. All cases assume elements stacked to infinite height and full water reflection. Voids within the fuel are either dry or flooded with water. The quantity "min r" is the minimum inside radius of the canister required to contain the arrangement.

Table 2. Effect of Chlorine on Reactivity of Flooded Fort Saint Vrain Fuel. The chlorine exists in the water used to flood the element but not in the reflecting water. An atom density of 30.0×10^{-4} atoms/b-cm corresponds to ~150 ppt of chlorine.

chlorine atom density (10^{-4} atoms/b-cm)		
	3 elements	7 elements
2	0.8864 ± 0.0034	1.1017 ± 0.0034
4	0.8337 ± 0.0031	1.0488 ± 0.0034
6	0.7864 ± 0.0029	0.9961 ± 0.0029
8	0.7471 ± 0.0033	0.9562 ± 0.0033
10	0.7251 ± 0.0034	0.9173 ± 0.0029
20	0.6001 ± 0.0031	0.7770 ± 0.0031
30	0.5221 ± 0.0026	0.6840 ± 0.0025

Figure 4. Effect of Chlorine on the Reactivity of Flooded Fort Saint Vrain Fuel Elements. The chlorine exists in the water used to flood the element but not in the reflecting water.

6.3 Shippingport PWR Core-2 Seed-2 (PWR)

PWR fuel is a "plate" type fuel that has zircaloy cladding and contains discrete burnable poison in the form of boron.^{17,18} The plates are comprised of a large number of fuel wafers which are sandwiched between sheets of cladding. There are 19 fuel plates per subassembly and four subassemblies per cluster as shown in Figure 5. The fuel wafers are made up of $\text{UO}_2\text{-ZrO}_2\text{-CaO}$. A PWR cluster has a beginning of life ^{235}U loading of 19.52 kg at an enrichment of 93.2%. Wafers are produced and loaded in a way that creates three zones within the cluster, each zone having a different fuel density.

The fuel matrix and surrounding cladding were all modelled discretely. Assumptions in the PWR cluster model include:

- ☞ End of life fuel loading information was used.
- ☞ Calcium was not included in the fuel.
- ☞ All four subassemblies were modelled with the zone loading of the most reactive subassembly.
- ☞ All four subassemblies had the same orientation. All fuel plates were parallel to one another.
- ☞ No boron was included.

Figure 6 shows the result of the first set of PWR calculations. Groups of one, two and four clusters stacked to an infinite height were modelled. Voids/coolant channels within the fuel were either dry or flooded with water. All cases assumed full water reflection, which conservatively envelopes the increase in reactivity due to canister and burial medium reflection. Four dry clusters (stacked to infinite height) only have a k_{eff} of 0.61, well within acceptable limits. Flooding with water, however, has a significant effect on reactivity. Two flooded PWR clusters have a reactivity in excess of critical. Flooding with pure water increased reactivity in excess of 40% in k_{eff} .

Once again flooding with brine solution instead of water was considered. Table 3 and Figure 7 show how, as was true for the Fort Saint Vrain calculations, reactivity is significantly reduced as the chlorine concentration increases. The reduction is not as dramatic as that observed for Fort Saint Vrain fuel elements. Four clusters flooded with a solution containing chlorine with an atom density of 3×10^{-3} atoms/b-cm has a k_{eff} of 0.98. This k_{eff} of 0.98, although significantly lower than the pure water flooded case, is still much more reactive than the dry case.

- water/void

- zircalloy/fuel plates

Figure 5. Cross Section of Typical Shippingport PWR Core 2 Seed 2 Cluster (PWR). Actual size is 7.378" x 7.378" (18.74 cm) with an effective fuel height of ~100" (254 cm).

Figure 6. Reactivity of PWR Clusters. All cases assume clusters are stacked to an infinite height and full water reflection. Voids within the fuel are either dry or flooded with water.

Table 3. Effect of Chlorine on Reactivity of Flooded PWR Clusters. The chlorine exists in the water used to flood the cluster but not in the reflecting water. An atom density of 30.0×10^{-4} atoms/b-cm corresponds to ~150ppt of chlorine.

	1 cluster	4 clusters
2	0.8376 ± 0.0026	1.2404 ± 0.0028
4	0.7977 ± 0.0025	1.2099 ± 0.0028
6	0.7685 ± 0.0024	1.1787 ± 0.0027
8	0.7443 ± 0.0026	1.1569 ± 0.0027
10	0.7248 ± 0.0023	1.1335 ± 0.0028
20	0.6421 ± 0.0021	1.0360 ± 0.0025
30	0.5888 ± 0.0022	0.9723 ± 0.0024

Figure 7. Effect of Chlorine on the Reactivity of Flooded Shippingport PWR Core-2 Seed-2. The chlorine exists in the water used to flood the cluster but not in the reflecting water.

7.0 Design Features (Active and Passive) and Administratively Controlled Limits and Requirements

Once fuel has been buried it may not be accessible and therefore administrative controls to prevent criticality after the fuel is buried may not be feasible.

Design features of a repository directly related to criticality safety are:

- ☛ Cannister material and dimensions
- ☛ Material and distance between neighboring fissile regions
- ☛ Repository location (this will directly affect the likelihood and magnitude of seismic and/or flooding events)
- ☛ Physical characteristics of the fuel

Results clearly indicate that burial of fuel in a medium containing large amounts of soluble chlorine (salt) will greatly reduce the reactivity of the fuel in the event of flooding. The chlorine also serves to neutronically isolate neighboring fissile regions. Guaranteeing the integrity of critical design features for the life of the repository will be a challenge.

8.0 Summary and Conclusions

8.1 Discussion of Results

Approximately 2 meters of separation between fissile regions is required to guarantee an effect on reactivity that is less than 1% in k_{eff} . This result is true for both tuff and salt. Since these results were based on the interaction of unmoderated infinite slabs of fissile material they are fairly conservative. When the design of the repository has progressed calculations with less conservative assumptions can be performed.

Stacked Fort Saint Vrain fuel elements and stacked PWR clusters have k_{eff} 's < 0.95 under the following conditions:

- 1- Fuel is arranged in a manner that is consistent with Figures 3 and 6. It is assumed that the fuel is intact.
- 2- The fuel is dry.

Fuel flooded with pure water is significantly more reactive than dry fuel. Only three Fort Saint Vrain fuel elements or one PWR cluster in the plane perpendicular to stacking could be allowed in the event of flooding with water and still maintain a $k_{eff} < 0.95$. Flooding with a brine solution is much less reactive than flooding with pure water. The chlorine in the brine solution acts as a fairly effective neutron poison, however benchmark experimental data will be required in order to take credit for chlorine present in the water.

A single PWR cluster containing ~15 kg of ^{235}U is only subcritical as long as the geometry of the fissile material is maintained.

8.2 General Remarks on the Prevention of a Criticality Accident

The issue of fuel degradation has not been addressed in this study. While the fuel remains intact fissile material is confined to a very specific geometrical configuration. Loss of the physical integrity of the fuel would result in the release and subsequent migration of fissile material. The

physical and chemical properties of uranium are such that it is reasonable to expect migrating uranium to collect in the same region. A significant increase in reactivity would then be expected. The loss of fuel integrity significantly affects the reactivity of the storage configurations discussed here. Once the integrity of highly enriched fuel is compromised, the critical mass of fuel material is greatly reduced. In fact, it could be argued that without guarantee of fuel integrity, a criticality could be postulated with as little as 760 grams ^{235}U . Realistically, several kilograms of ^{235}U may be required to overcome the effects on reactivity due to an imperfect geometry and the presence of neutron absorbing materials. Taking credit for these geometry and material effects in a safety analysis would have to be justified. The storage of highly enriched ^{235}U in quantities significantly greater than 760 grams for an indeterminate time could be difficult to defend unless the risk of a criticality is deemed acceptable.

8.3 Mechanisms of a Criticality Accident

Characteristics of a criticality include high radiation fields, the production of fission products, and the generation of energy in the form of heat. Mechanisms that may decrease the reactivity of an already supercritical system include

- ☞ disassembly
- ☞ thermal expansion
- ☞ doppler broadening of ^{238}U resonances (generally not significant in highly enriched fuels)
- ☞ loss of moderator (typically through evaporation)
- ☞ bubble formation in moderator (from boiling or fission product gases)
- ☞ depletion of fissile material

All of these shutdown mechanisms except the depletion of fissile material should be assumed to be reversible in a geologic repository. Given that a criticality occurs, it is reasonable to expect it will reoccur until something permanent takes place to stop it. For example, if an increase in temperature causes an already critical system to become subcritical it is reasonable to expect that when the system cools it will once again become critical.

Direct radiation fields (primarily neutron and gamma) would have no significant consequence in a geologic repository. Several meters of either tuff or salt is all that would be required to completely localize the effects of neutron and gamma fields. The production of fission products is of no consequence as long as the fission products remain contained. Presumably long term geologic storage of spent nuclear fuel would be designed to contain fission products to within acceptable limits. Another issue to be considered is migration of fission products in an environment that is generating large amounts of heat.

The average fission releases approximately 3.2×10^{-11} joules of energy. Roughly 80% of this energy is in the form of fission product kinetic energy and therefore heat. Assuming an initial burst of 10^{17} (typically between 10^{16} and 10^{20}) fissions takes place, the total heat generated would

be $\sim 2.6 \times 10^6$ joules. This energy release can be assumed to take place on a fairly short timescale (from milliseconds to seconds). The effect that this type of power output would have on the migration of fission products must be considered.

Additional issues that require consideration include:

- ☛ The scenario presented above is by no means conservative, the only permanent shutdown mechanism for a critical system is depletion of the fissile isotope. A criticality event with many orders of magnitude more fissions and lasting a much longer time can be postulated.
- ☛ Buried spent fuel can be expected to have a large inventory of long lived fission products. The migration of these fission products in the event of a criticality must be considered. A criticality event will also produce potentially large quantities of short lived fission products. These short lived fission products may have different transport properties than the long lived fission products. In particular, a large short lived gaseous fission product inventory can be expected.

9.0 References

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Appendix A: Materials and Compositions

Salt

Salt was assumed to be pure NaCl with a density of 2.2 grams/cc, giving:

<u>Material</u>	<u>Number Density (atoms/b-cm)</u>
Sodium	2.267×10^{-2}
Chlorine	2.267×10^{-2}

Tuff

Tuff was assumed to have a density of 2.2 grams/cc and the following constituents:¹⁹

<u>Material</u>	<u>wt%</u>	<u>Number Density (molecules/b-cm)</u>
SiO ₂	69.7	1.537×10^{-2}
Al ₂ O ₃	15.6	2.027×10^{-3}
K ₂ O	6.6	9.282×10^{-4}
Na ₂ O	4.1	8.761×10^{-4}

Total=96%

Other constituents were ignored.

Fort Saint Vrain

Volume of hexagonal element = $0.8660 (D^2) (L)$ where D is the distance across the flats and L is the length of the element. For Fort Saint Vrain D=35.9969 cm and L=79.2988 cm giving a volume of 88985 cm³.

For the Fort Saint Vrain discrete model:

210 fuel rod holes contain homogeneous mixtures of all the silicon, thorium, ²³⁵U and ²³³U. The total volume of the holes is:

$$Volume = 210 rods \times ((0.635 cm)^2) \times 3.1415 \times 79.2988 cm = 2.110 \times 10^4 cm^3$$

giving;

<u>Material</u>	<u>Mass(kg)</u>	<u>Number Density (atoms/b-cm)</u>
Silicon	4	4.066×10^{-3}
Thorium	13	1.600×10^{-3}
²³⁵ U	0.81	9.839×10^{-5}
²³³ U	0.27	3.308×10^{-5}

The carbon is assumed to be evenly distributed everywhere except in the void/coolant channels:

$$\text{Carbon Volume} = (\text{total volume}) \times (1.0 - \text{void volume fraction}) = 88985 \text{ cm}^3 \times 0.8004 = 7.122 \times 10^4 \text{ cm}^3$$

Assuming 110 kg of carbon, one gets a carbon atom density of 7.751×10^{-2} atoms/b-cm.

Shippingport PWR Core 2 Seed 2

<u>Material</u>	<u>Number Density (atoms/b-cm)</u>
Zone 1 ^{235}U	4.7509×10^{-3}
Zone 1 ^{238}U	1.1100×10^{-3}
Zone 1 Oxygen	5.7034×10^{-2}
Zone 1 Zirconium	2.2656×10^{-2}
Zone 2 ^{235}U	3.1557×10^{-3}
Zone 2 ^{238}U	7.3087×10^{-4}
Zone 2 Oxygen	5.7906×10^{-2}
Zone 2 Zirconium	2.5067×10^{-2}
Zone 3 ^{235}U	1.9516×10^{-3}
Zone 3 ^{238}U	4.5204×10^{-4}
Zone 3 Oxygen	5.8562×10^{-2}
Zone 3 Zirconium	2.6877×10^{-2}
Zirconium cladding	2.2656×10^{-2}

Other

Material in pure water:

<u>Material</u>	<u>Number Density (atoms/b-cm)</u>
Hydrogen	6.691×10^{-2}
Oxygen	3.346×10^{-2}

Typical chlorine density in a brine solution:

Assuming the solution has a density of 1.2 g/cc and the chlorine concentration is 150 ppt then

$$\rho_{\text{Cl}} \left(\frac{\text{g}}{\text{cc}} \right) = \text{ppt} \left(\frac{\text{g (Cl)}}{\text{kg (sol)}} \right) \cdot \left(\frac{\text{kg (sol)}}{1000 \text{ g (sol)}} \right) \cdot \rho_{\text{sol}} \left(\frac{\text{g}}{\text{cc}} \right) = 0.18 \left(\frac{\text{g}}{\text{cc}} \right) \Rightarrow 3.057 \times 10^{-3} \left(\frac{\text{atoms}}{\text{b-cm}} \right)$$

Appendix B: Typical Input Listings

Infinite Slabs

```
1 0 0
Isolation in salt
/ Block 1
igeom=slab ngroup=16 isn=6 niso=2 mt=2 nzone=2 im=6 it=1000
maxscm=50000 maxlcm=200000 t

/ Block 2
xmesh= 0 10 20 21 30 130
xints= 150 150 200 200 150 150
zones=1 1 1 2 2 2 t

/ Block 3
lib=f16 writmxs=macbcd
chivec= .204 .344 .168 .180 .090 .014 10z
maxord=1 ihm=22 iht=3 ihs=7 ititl=1 ifido=2 i2lp1=1 t

/ Block 4
matls=isos assign=matls t

/ Block 5
ievt=1 isct=1 ith=0 ibl=1 ibr=1 epsi=1.-5 t

/ Block 6
zned=1 t
```

Fort Saint Vrain

FORT ST. VRAIN THREE FLOODED ELEMENTS

```
1 3 -1 -1 u=1 $coolant channel
2 1 8.331-2 -2 u=1 $2-7 are fuel channels
3 1 8.331-2 -3 u=1
4 1 8.331-2 -4 u=1
5 1 8.331-2 -5 u=1
6 1 8.331-2 -6 u=1
7 1 8.331-2 -7 u=1
8 2 7.751-2 +1 +2 +3 +4 +5 +6 +7 u=1 $graphite
9 0 -8 +9 -10 +11 -12 +13 u=2 lat=2 fill=1 $unit cell boundary of lattice
10 0 (+30:-31:-32:-33:+34:-35)
(#26 #27 #28 #29 #30 #31)
-14 +15 -16 +17 -18 +19 u=3 fill=2 $boundary of lattice
```

```

11 2 7.751-2 (+14:-15:+16:-17:+18:-19) -50 u=3 $outer boundary of FHU
12 3 -1 -28 u=3 $center void
13 3 -1 -29 trcl=(0 +3.25628 0) u=3 $13-24 fuel/coolant channels
14 3 -1 -29 trcl=(0 -3.25628 0) u=3
15 1 8.331-2 -29 trcl=(+3.7592 0 0) u=3
16 1 8.331-2 -29 trcl=(-3.7592 0 0) u=3
17 1 8.331-2 -29 trcl=(+1.8796 +3.25628 0) u=3
18 1 8.331-2 -29 trcl=(-1.8796 +3.25628 0) u=3
19 1 8.331-2 -29 trcl=(+1.8796 -3.25628 0) u=3
20 1 8.331-2 -29 trcl=(-1.8796 -3.25628 0) u=3
21 3 -1 -29 trcl=(+2.8194 +1.62814 0) u=3
22 3 -1 -29 trcl=(-2.8194 +1.62814 0) u=3
23 3 -1 -29 trcl=(+2.8194 -1.62814 0) u=3
24 3 -1 -29 trcl=(-2.8194 -1.62814 0) u=3
25 2 7.751-2 (#12 #13 #14 #15 #16 #17 #18 #19 #20 #21 #22 #23 #24)
    -30 +31 -32 +33 -34 +35 u=3 $graphite
26 2 7.751-2 -29 trcl=(+9.398 +16.27886 0) u=3 $poison rod
27 2 7.751-2 -29 trcl=(-9.398 -16.27886 0) u=3 $poison rod
28 3 -1 -29 trcl=(-9.398 +16.27886 0) u=3 $28-31 void channels
29 3 -1 -29 trcl=(+9.398 -16.27886 0) u=3
30 3 -1 -29 trcl=(+18.796 0 0) u=3
31 3 -1 -29 trcl=(-18.796 0 0) u=3
32 3 -1.0 -50 u=4 $alternate universe
33 0 -20 +21 -22 +23 -24 +25 lat=2 u=5 $lattice of FHUs
    fill=-2:2 -2:2 0:0 4 4 4 4 4
        4 4 3 4 4 4 3 3 4 4
        4 4 4 4 4 4 4 4 4 4
34 0 -51 +27 -26 fill=5
35 0 +51:-26:-27 $zero importance outside world

1 cz +0.79375 $coolant channel
2 c/z +1.8796 +0.0 0.635 $2-7 are fuel channels
3 c/z +0.9398 +1.627781352 0.635
4 c/z -0.9398 +1.627781352 0.635
5 c/z -1.8796 +0.0 0.635
6 c/z -0.9398 -1.627781352 0.635
7 c/z +0.9398 -1.627781352 0.635
8 p +1.7320508 1 0 +3.255562704 $8-13 are unit cell boundary of lattice
9 p +1.7320508 1 0 -3.255562704
10 py +1.627781352
11 py -1.627781352
12 p -1.7320508 1 0 +3.255562704
13 p -1.7320508 1 0 -3.255562704
14 p +1.7320508 1 0 +34.14522 $14-19 are boundary of lattice
15 p +1.7320508 1 0 -34.14522
16 py +17.07261

```

```

17 py -17.07261
18 p -1.7320508 1 0 +34.14522
19 p -1.7320508 1 0 -34.14522
20 p +1.7320508 1 0 +35.99688      $20-25 are outer boundary of FHU
21 p +1.7320508 1 0 -35.99688
22 py +17.99844
23 py -17.99844
24 p -1.7320508 1 0 +35.99688
25 p -1.7320508 1 0 -35.99688
26* pz +39.6494
27* pz -39.6494
28 cz +1.74625
29 cz +0.635
30 p +1.7320508 1 0 +8.18134      $30-35 are boundary of inner hexagon
31 p +1.7320508 1 0 -8.18134
32 py +4.09067
33 py -4.09067
34 p -1.7320508 1 0 +8.18134
35 p -1.7320508 1 0 -8.18134
50 cz +40.0
51 cz +70.0

```

```

mode n
imp:n 1 33r 0
m1 92235.50c 9.839-5 92233.50c 3.308-5 90232.50c 1.600-3
    14000.50c 4.066-3 6012.50c 7.751-2
mt1 grph.01t
m2 6012.50c 7.751-2
mt2 grph.01t
m3 1001.50c 2 8016.50c 1
mt3 lwtr.01t
kcode 1000 1.0 5 55
ksrc 9.398 0 0
print

```

Shippingport PWR-Core 2 Seed 2

```

=CSAS25
BURYIT
27G INF
U-235      1 0 4.7509-03 END
U-238      1 0 1.1100-03 END
O          1 0 5.7034-02 END
ZR         1 0 2.2656-02 END
U-235      2 0 3.1557-03 END
U-238      2 0 7.3087-04 END

```

O 2 0 5.7906-02 END
 ZR 2 0 2.5067-02 END
 U-235 3 0 1.9516-03 END
 U-238 3 0 4.5204-04 END
 O 3 0 5.8562-02 END
 ZR 3 0 2.6877-02 END
 ZR 4 0 2.2656-02 END
 H2O 5 1 END
 CL 5 2-4 END
 H2O 6 1 END
 END COMP
 KENOV.A ONE PWR SEED2 EOL CLUSTER
 READ PARAM TME=25. PLT=YES RUN=YES GEN=250 NPG=500 END PARAM
 READ GEOMETRY
 UNIT 1
 CUBOID 2 1 .0457 -.0457 2.50825 1.0719 246.0 0.0
 CUBOID 3 1 .0457 -.0457 7.6657 1.0719 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 2
 CUBOID 2 1 .0457 -.0457 3.97637 .9093 246.0 0.0
 CUBOID 3 1 .0457 -.0457 7.8283 .9093 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 3
 CUBOID 2 1 .0457 -.0457 5.83184 .75692 246.0 0.0
 CUBOID 3 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 4
 CUBOID 2 1 .0457 -.0457 6.56336 .75692 246.0 0.0
 CUBOID 3 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 5
 CUBOID 2 1 .0457 -.0457 1.44272 .75692 246.0 0.0
 CUBOID 1 1 .0457 -.0457 5.10032 .75692 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.29488 .75692 246.0 0.0
 CUBOID 3 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 6
 CUBOID 5 1 .0927 -.0927 3.8976 -3.8976 246.0 0.0
 CUBOID 4 1 .0927 -.0927 4.3688 -4.3688 246.0 0.0
 UNIT 7
 CUBOID 5 1 .1020 -.1020 3.9992 -3.9992 246.0 0.0
 CUBOID 4 1 .1020 -.1020 4.3688 -4.3688 246.0 0.0
 UNIT 8
 CUBOID 5 1 .1020 -.1020 4.0246 -4.0246 246.0 0.0
 CUBOID 4 1 .1020 -.1020 4.3688 -4.3688 246.0 0.0
 UNIT 9

CUBOID 5 1 .0984 -.0984 4.0246 -4.0246 246.0 0.0
 CUBOID 4 1 .0984 -.0984 4.3688 -4.3688 246.0 0.0
 UNIT 10
 CUBOID 5 1 .1020 -.1020 3.8976 -3.8976 246.0 0.0
 CUBOID 4 1 .1020 -.1020 4.3688 -4.3688 246.0 0.0
 UNIT 11
 CUBOID 4 1 .2574 -.2574 4.3688 -4.3688 246.0 0.0
 UNIT 12
 CUBOID 3 1 .0457 -.0457 6.22935 1.0719 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.6657 1.0719 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 13
 CUBOID 3 1 .0457 -.0457 4.76123 .9093 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.8283 .9093 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 14
 CUBOID 3 1 .0457 -.0457 2.90576 .75692 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 15
 CUBOID 3 1 .0457 -.0457 2.17424 .75692 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 16
 CUBOID 3 1 .0457 -.0457 1.44272 .75692 246.0 0.0
 CUBOID 2 1 .0457 -.0457 3.63728 .75692 246.0 0.0
 CUBOID 1 1 .0457 -.0457 7.29488 .75692 246.0 0.0
 CUBOID 2 1 .0457 -.0457 7.98068 .75692 246.0 0.0
 CUBOID 4 1 .0978 -.0978 8.7376 0.0 246.0 0.0
 UNIT 17
 CUBOID 5 1 .0978 -.0978 1.27 0.0 246.0 0.0
 UNIT 18
 CUBOID 5 1 .0927 -.0927 1.27 0.0 246.0 0.0
 UNIT 19
 CUBOID 5 1 .1020 -.1020 1.27 0.0 246.0 0.0
 UNIT 20
 CUBOID 5 1 .0984 -.0984 1.27 0.0 246.0 0.0
 UNIT 21
 CUBOID 5 1 .2574 -.2574 1.27 0.0 246.0 0.0
 UNIT 22
 CUBOID 5 1 1.27 0.0 1.27 0.0 246.0 0.0
 UNIT 23
 CUBOID 5 1 1.27 0.0 8.7376 0.0 246.0 0.0
 GLOBAL
 ARRAY 1 3*0
 REFLECTOR 6 2 3 0 3 0 0 0 10

```
END GEOM
READ BIAS ID=500 2 11 END BIAS
READ BOUNDS
ZFC=PERIODIC
-YB=MIRROR
-XB=MIRROR
END BOUNDS
READ ARRAY
ARA=1 NUX=83 NUY=3 NUZ=1
LOOP
11 1 41 40 1 3 1 3*1
6 2 40 38 1 3 1 3*1
1 3 39 36 1 3 1 3*1
10 4 38 34 1 3 1 3*1
2 5 7 2 1 3 1 3*1
2 35 37 2 1 3 1 3*1
7 6 36 30 1 3 1 3*1
8 8 34 2 1 3 1 3*1
5 9 25 2 1 3 1 3*1
9 14 20 2 1 3 1 3*1
9 28 32 2 1 3 1 3*1
4 27 29 2 1 3 1 3*1
3 31 33 2 1 3 1 3*1
23 42 42 1 1 3 1 3*1
22 42 42 2 2 2 1 3*1
11 43 83 40 1 3 1 3*1
6 44 82 38 1 3 1 3*1
1 45 81 36 1 3 1 3*1
10 46 80 34 1 3 1 3*1
2 47 49 2 1 3 1 3*1
2 77 79 2 1 3 1 3*1
8 50 76 2 1 3 1 3*1
7 48 78 30 1 3 1 3*1
3 51 53 2 1 3 1 3*1
9 52 56 2 1 3 1 3*1
9 64 70 2 1 3 1 3*1
4 55 57 2 1 3 1 3*1
5 59 75 2 1 3 1 3*1
21 1 41 40 2 2 1 3*1
21 43 83 40 2 2 1 3*1
18 2 40 38 2 2 1 3*1
18 44 82 38 2 2 1 3*1
17 3 39 2 2 2 1 3*1
17 45 81 2 2 2 1 3*1
19 4 38 2 2 2 1 3*1
19 46 80 2 2 2 1 3*1
```

```
20 14 20 2 2 2 1 3*1
20 28 32 2 2 2 1 3*1
20 52 56 2 2 2 1 3*1
20 64 70 2 2 2 1 3*1
12 3 39 36 3 3 1 3*1
12 45 81 36 3 3 1 3*1
13 5 7 2 3 3 1 3*1
13 35 37 2 3 3 1 3*1
13 47 49 2 3 3 1 3*1
13 77 79 2 3 3 1 3*1
16 9 25 2 3 3 1 3*1
16 59 75 2 3 3 1 3*1
15 27 29 2 3 3 1 3*1
15 55 57 2 3 3 1 3*1
14 31 33 2 3 3 1 3*1
14 51 53 2 3 3 1 3*1
END LOOP
END ARRAY
READ PLOT
TTL=* X-Y OF PWR CLUSTER *
NCH=* 123Z.* PIC=MIX PLT=YES
XUL=0. YUL=10. ZUL=20.
XLR=10. YLR=0. ZUL=20.
UAX=1 VDN=-1 NAX=100 END
END PLOT
END DATA
END
```

END

DATE
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