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Criticality Safety Study of the MSRE Auxiliary Charcoal Bed

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1. INTRODUCTION

The Molten Salt Reactor Experiment (MSRE) was operated from June 1965 to December 1969. The objective of the experiment was to investigate the practicality of developing a power reactor consisting of a graphite lattice with circulating molten uranium salt as fuel for application in central power stations. When the experiment was terminated in 1969, approximately 4710 kg of salt containing approximately 36.3 kg of uranium, 675 g of plutonium, and various fission products were transferred to two fuel drain tanks (FDTs). The almost 30.5 kg of ^{233}U in the salt is the primary fissile constituent, but about 0.93 kg of ^{235}U is also present.

After the fuel was transferred to the FDTs, it was allowed to solidify. The solidified salt was reheated (while remaining below the melting point) on a yearly basis to recombine fluorine released by radiolytic decomposition. Because of a possible radioactive contamination problem, the yearly reheating was discontinued after the 1989 reheat (performed in January 1990) in order to investigate the source of the problem.

Following a year-long effort, an upgraded nuclear criticality safety (NCS) approval for the storage of MSRE fuel in the FDTs was completed at the end of 1993. Based on information supplied by MSRE facility personnel and reviews of engineering drawings and facility documentation it was assumed that (1) all of the salt, fissile material, and fission products transferred to the FDTs were still in the FDTs; (2) there were no open pathways for material to either leave or enter the FDTs; and (3) there was no water currently in the system and no pathways for water to enter the system. The set of k_{eff} calculations that support the NCS analysis of MSRE fuel storage in the FDTs was made using these assumptions.[1]

In April 1994 a gas sample from the MSRE off-gas system (OGS) indicated that uranium had migrated from the FDTs into the OGS. Further investigation revealed a likely accumulation of approximately 2.6 kg of uranium in the auxiliary charcoal bed (ACB), which is located in the concrete-lined charcoal bed cell (CBC) below ground level outside the MSRE building.[2] The NCS situation was further complicated by the CBC being filled with water up to the overflow pipe, which completely submerged the ACB. Thus there was not only an increased risk of criticality because of water reflection in the ACB, but also because of potential moderation in the ACB in case of water inleakage. Leakage into the ACB would result in a direct path for water between the CBC and the OGS or FDTs, thus increasing the risk of criticality in these areas.

When uranium was discovered in the ACB, a number of steps were immediately taken to try to understand and ameliorate the situation. An analysis of the as-found ACB configuration indicated there might be enough uranium present to achieve criticality if water leaked into the volume of the ACB containing the uranium. The analysis also showed that if the remainder of the uranium in the FDTs were also to migrate into the submerged ACB, subcriticality could not be ensured even without water in-leakage. This analysis used the 27-group ENDF/B-IV cross-section set. Previous studies have shown that this cross-section set consistently produces the most conservative results.[1] However, without water reflection, the limited diameter of the pipe is sufficient to ensure subcriticality even if a water leak occurred. The first action taken was to immediately lower the water level in the CBC to below the level of the pipe connection to the OGS. After an analysis to determine thermal

conditions in the absence of water cooling, the water level was further lowered to 2 ft beneath the bottom of the inferred uranium accumulation. The ACB pit was later completely drained of water.

After all the actions were completed, a validation of the results obtained for the ACB was performed. A set of 24 experiments, containing ^{233}U as the primary fissile component, were calculated using both the 27-group ENDF/B-IV and 238-group ENDF/B-V cross-section sets. An analysis of these calculations showed that the 238-group cross sections produced better results. The ACB cases that represented the moderation extremes were therefore recalculated using the 238-group cross sections. These results are discussed at the end of this paper.

Because the equipment and facilities described in this report are over 30 years old, all reports, drawings, and schematics used SI units. To maintain continuity with the legacy documentation, English units are used when describing the MSRE facilities. SI units are employed whenever new calculations, such as material masses and densities, are done.

2. CHARCOAL BED MODEL

To provide a basis for any actions to be taken after the uranium migration into the ACB was discovered, an analysis was performed to determine the amount of uranium present in the ACB, how the uranium was distributed, and the NCS consequences. Estimates of the uranium content and its distribution profile were made based on analyses of the measured gamma radiation and temperature profiles in the vicinity of the ACB. From the analysis, the most probable configuration appears to be a relatively uniform distribution of 2.6 kg of uranium in the top 12 in. of the activated charcoal that fills the ACB.[2]

The ACB is fabricated from 6-in. Sch. 10 stainless steel pipe. At the inlet end of the ACB, starting from the top, the pipe contains a 2-in. air gap, a 2.5-in. section of steel wool, and from that point down, activated charcoal. Two 1/16-in.-thick stainless steel perforated disks enclose the steel wool, one on each end. A 1/8-in. Sch. 40 stainless steel thermocouple pipe extends from the top of the ACB to 6 in. below the top of the charcoal. The base case for the k_{eff} calculations assumes that all the uranium, in the form of UF_4, is in the top 12 in. of the charcoal. A schematic of the base ACB model used in the criticality calculations is shown in Fig. 1.

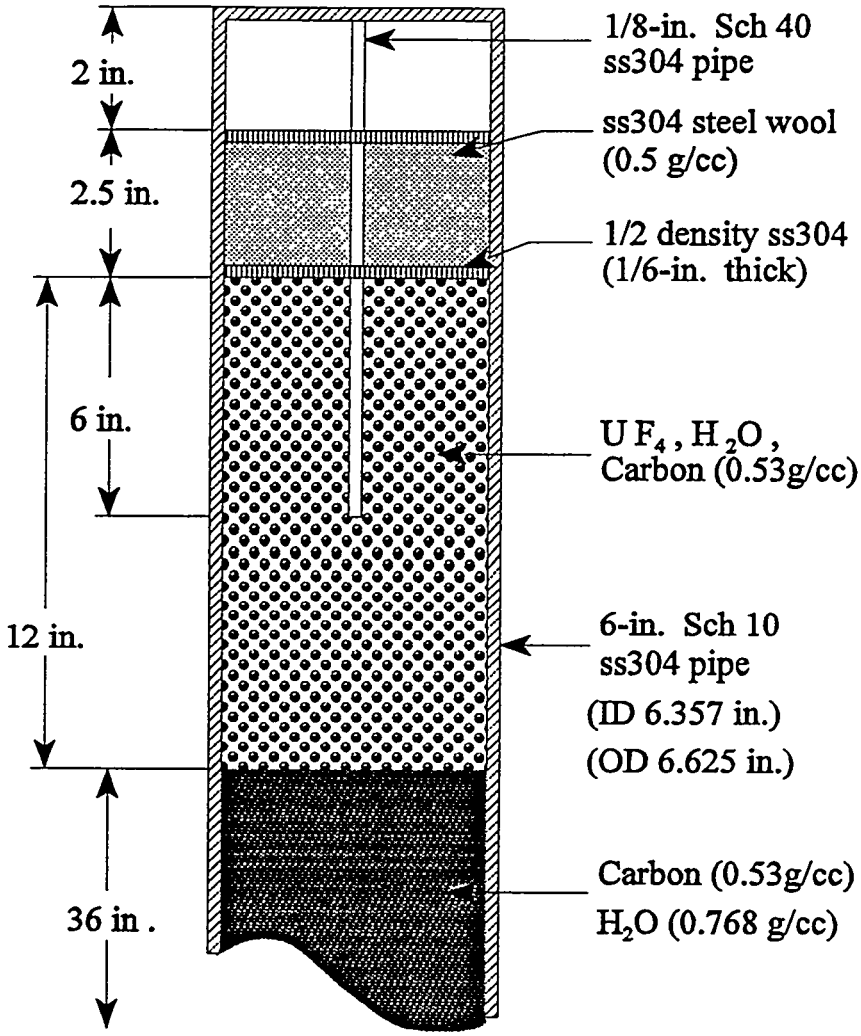

Fig. 1. KENO-V.a model of the MSRE ACB.

3. CALCULATIONS AND ANALYSIS

Initially several unknowns needed to be addressed concerning the criticality safety of the ACB. The quantity and distribution of uranium in the ACB were unknown and needed to be estimated. The best estimate from the data is approximately 2.6 kg of ^{233}U in the top 12 in. of the activated charcoal. Since this is only an estimate, a range of values around the estimate needs to be examined for criticality consequences. Most of the calculations were performed using 2.0 kg of ^{233}U, an earlier estimate. The additional 0.6 kg of ^{233}U has little reactivity effect for the conditions of the calculations reported here, as is later discussed and shown in Table 2.

Table 1 contains the results of a k_{eff} study [$k_{eff} \pm 1$ one standard deviation (σ)] over a range of UF_4 deposition lengths. It would be expected that the uranium in the ACB would have the same isotopic distribution as that in the FDTs. The 2.0 kg of ^{233}U is distributed uniformly over lengths ranging from 5 to 24 in. Each distribution is evaluated for two extremes: (1) the ACB contains no water and (2) the ACB is internally flooded. All cases are radially reflected by 3 feet of water.

All calculations in this section were done using the 27-group ENDF/B-IV cross-section library. This library was chosen because it produced the most conservative results in the 1993 analysis of the FDTs.[1]

Table 1. Summary of k_{eff} values for the ACB containing 2.0 kg of ^{233}U

UF_4 deposition length (in.)	Dry k_{eff} ($\pm\sigma$)	Internally flooded k_{eff} ($\pm\sigma$)
5.0	0.3624 (0.0034)	0.7879 (0.0062)
8.0	0.3770 (0.0036)	0.8838 (0.0059)
11.0	0.3823 (0.0041)	0.9264 (0.0062)
12.0	0.3905 (0.0018)	0.9408 (0.0031)
17.0	0.3903 (0.0039)	0.9645 (0.0057)
24.0	0.3875 (0.0040)	0.9462 (0.0056)

The results in Table 1 show that as long as ^{233}U is not moderated by water, the ACB remains far subcritical and there is no criticality safety concern. If water were to leak into the ACB and moderate the UF_4-C mixture, there is significant potential for criticality if the UF_4 deposition length exceeds 11 in. The reactivity of the system peaks with a deposition length between 12 and 24 in. Since the most likely deposition length lies within the range of greatest reactivity, it is imperative to prevent water from internally flooding the ACB.

The uncertainty associated with the amount of uranium present in the ACB must also be addressed. Table 2 contains the results of an analysis where the UF_4 deposition length is kept constant while the UF_4 mass is varied by ±50%. In all cases, the uranium in the ACB is assumed to have the same isotopic distribution as that in the FDTs and is deposited uniformly over the top 12 in. of the ACB. As in the other calculations, a 3-ft-thick water reflector surrounds the pipe.

Table 2. Summary of k_{eff} values for the ACB containing various masses of ^{233}U with a fixed UF_4 deposition length

^{233}U mass (kg)	UF_4 deposition length (in.)	Dry k_{eff} ($\pm\sigma$)	Flooded k_{eff} ($\pm\sigma$)
1.5	12.0	0.3636 (0.0016)	0.9167 (0.0026)
2.0	12.0	0.3905 (0.0018)	0.9408 (0.0031)
2.5	12.0	0.4074 (0.0020)	0.9480 (0.0031)
3.0	12.0	0.4289 (0.0021)	0.9554 (0.0031)

The results of Table 2 also show that, for the range of ^{233}U mass considered in Table 1, as long as the section of the ACB containing uranium is not water-moderated, there is no criticality safety concern. If water internally floods the ACB, there is a significant potential for criticality if the UF_4 deposit contains 2.0 kg or more of ^{233}U. Over the range of ^{233}U mass examined, the reactivity of the system increases as more ^{233}U accumulates in the ACB. From Table 2 it is also seen that k_{eff} increases only in the third decimal place as the ^{233}U increases from 2.0 to 2.5 kg.

The effects on system reactivity of other conditions associated with UF_6 migration to the ACB were also examined. These include the effect of a nonlinear UF_4 deposition in the ACB, the reactivity worth of the reflector, and the reactivity worth of the carbon mixed with the UF_4. Table 3 summarizes the results of these examinations.

In case 1 of Table 3, the 2.0 kg of ^{233}U (in UF_4), is distributed linearly in the top 12 in. of the activated charcoal in the ACB. The 12-in. deposition length is divided into 12 1-in. sections. The uranium density varies linearly in these sections so that the ratio of the uranium density between the bottom and top section is 2.0. Within statistical uncertainty, the reactivity of this case is the same as that for the case with the uranium uniformly distributed.

Case 2 is for a different nonuniform uranium distribution in the top 12 in. of the activated charcoal of the ACB. The 12-in. deposition length is divided into 6 thin (~0.1-in.) layers of C + UF_4 separated by layers of C + H_2O. The reactivity of this case decreased by about 7% compared with the identical case with the uranium uniformly distributed over 12 in. The UF_4 is far undermoderated, and the ACB pipe has a small enough diameter that any separation of the UF_4 into clumps decreases the reactivity of the system.

Table 3. Summary of k_{eff} values for other conditions in the ACB

Case	^{233}U mass (kg)	UF_4 deposition length (in.)	k_{eff} ($\pm\sigma$)	Description
1	2.0	12.0	0.9390 (0.0031)	Flooded, reflected, UF_4 linear gradient
2	2.0	12.0	0.8585 (0.0029)	Flooded, reflected, stratified H_2O and UF_4
3	2.0	12.0	0.9366 (0.0027)	Flooded, reflected, no thermocouple pipe
4	2.0	12.0	0.6522 (0.0026)	Flooded, unreflected
5	2.0	12.0	0.7707 (0.0028)	Flooded, unreflected, no C in UF_4
6	2.0	12.0	1.0299 (0.0028)	Flooded, reflected, no C in UF_4
7	20.5	12.0	0.8911 (0.0026)	Dry, reflected, as much uranium as possible in the top 12 in. of the ACB
8	30.5	18.0	0.9491 (0.0021)	Dry, reflected, all available uranium in the top 18 in. of the ACB

In case 3, the worth of the thermocouple pipe through the top 6 in. of the activated charcoal is examined. For this case, the thermocouple pipe is removed from the model and the densities of the uranium and water modified to account for the increased volume. The reactivity of case 3 is the same within statistical uncertainty as that for the 12-in. base case from Table 1.

In case 4, the reactivity worth of the water reflector is examined. In this case, the UF_4-C mixture is water-moderated but the water reflector is removed. As seen from the result, without the water reflector surrounding the ACB, criticality is no longer possible.

Cases 5 and 6 assume that the activated charcoal bed has settled enough so that 12 in. of a uniform UF_4-water mixture could accumulate between the top of the activated charcoal and the perforated disc below the steel wool. The only difference between cases 5 and 6 is the presence of a reflector. Case 5, which assumes no water reflector, is far subcritical. Case 6, which assumes a 3-ft water reflector on all sides, is significantly supercritical. For this configuration the addition of the reflector increases the k_{eff} of the system by about 25%.

In cases 7 and 8, as much UF_4 as the interstitial space will allow while maintaining the charcoal density is accumulated in the top of the ACB. The ACB is assumed to be dry and surrounded by a 3-ft water reflector. Case 7 uniformly distributes approximately 20.5 kg of ^{233}U and the other associated isotopes of uranium, in the form of UF_4, in the top 12 in. of the activated charcoal. This amount is limited by the density of the charcoal and the UF_4. Even with this amount of uranium in the ACB, criticality is not possible as long as water is not present. In case 8, all the

uranium, ~30.5 kg of ^{233}U, is uniformly distributed in the top 18 in. of the activated charcoal. Again, no water is present in the system but there is a 3-ft water reflector. Criticality is certainly possible in this case, considering the extent of the uncertainties. However, it is considered essentially incredible that all the uranium would migrate and then concentrate in the top of the ACB.

This preliminary analysis of the ACB has shown that there are no criticality safety concerns unless water mixes with the uranium in the ACB and the ACB is water reflected. However, considering the uncertainty associated with the ^{233}U cross sections used to perform the calculations, as well as the questions concerning the amount and distribution profile of the uranium in the ACB, the presence of significant amounts of water in the ACB pit would be a serious criticality safety concern.

4. VERIFICATION AND VALIDATION

An NCS analysis requires that calculational analysis of a system be performed on a platform that has been verified and validated for the type of system being evaluated. Unfortunately, no such platform currently exists for a system where ^{233}U makes up the majority of the fissile material and the average fission energy is in the epithermal range as in the ACB. All calculations in Sect. 3 were performed on CA02.CAD.ORNL.GOV, which is an IBM RISC-6000 Model 580 workstation, using a controlled version of SCALE. Although controlled, this version of SCALE does not meet all the QA requirements of a properly verified and validated platform.

Since a properly verified and validated QA-controlled platform does not exist for the ACB, an attempt was made to verify and validate an existing platform. The ORNL NCS section uses XNCSE1.ORFS.ORNL.GOV, which is a DEC ALPHA workstation, to perform criticality calculations. Although the version of SCALE used on the ALPHA has not been verified and validated, it is maintained under strict QA control. All the test cases used to verify that the code has been properly installed have been run satisfactorily. These cases contain all the geometry configurations found in the ACB. Based on the successful completion of all SCALE test problems, it is arguably correct that this platform is verified and validated for the geometric configurations under consideration.

The remaining problems associated with producing a properly verified and validated system stem from the materials present in the ACB. There are critical experiments where ^{233}U is the primary fissile material. A compilation of critical experiments, including some where ^{233}U in the primary fissile material was prepared under the direction of B. L. Koponen at Lawrence Livermore Laboratory.[3] Unfortunately, these experiments have either thermal or fast energy spectrum, whereas the systems mocking up the ACB are primarily epithermal systems. A set of 24 experiments that would best represent the ACB were selected and analyzed using the SCALE system on the DEC ALPHA.

Validation of the platform used to analyze the ACB utilizes a set of 24 critical experiments taken from an ORNL document produced to assist in the validation of a platform for use in analyzing the Building 3019 storage wells.[3] These experiments are referenced using the Koponen citation number, which refers to a compilation of critical experiments prepared under the direction of B. L. Koponen at Lawrence Livermore Laboratory.[4] Table 4 contains a list of these experiments and a brief description of their characteristics. The experiments were analyzed using the 27-group ENDF/B-IV cross-section library and the 238-group ENDF/B-V cross-section library. Tables 5 and 6 contain the result of these experiments for the two libraries, respectively.

The 24 experiments chosen for analysis can be divided into three categories. The first category consists of the ten fast experiments listed in Table 4 having the Koponen citation numbers 1727-2. They consist of highly enriched ^{233}U metal spheres surrounded by either a void or a metal reflector.[5] The highly enriched ^{233}U metal spheres in experiments 1727-01 to -08 are surrounded by metal reflectors. Experiments 1727-14 and -15 consist of highly enriched ^{233}U metal spheres surrounded by a highly enriched ^{235}U metal shell. As shown in Table 5 for these cases, the 27-group library always calculates subcritical, down to 4% below critical. As shown in Table 6, seven of the ten cases calculate critical within 2σ using the 238-group library. The maximum deviation from critical, which occurs for the case with no reflector, is less than 1% low for the 238-group library.

The second category of experiments consists of the four thermal experiments listed in Table 4 having the Koponen citation numbers 1211-**, where the ** are any pair of digits from Table 5. They consist of arrays composed of uranyl nitrate solutions of different concentrations in stainless steel cans surrounded by a polyethylene reflector.[6] The highly enriched ^{233}U uranyl nitrate solutions are arranged in either a 2 × 2 × 2 or a 3 × 3 × 3 array. As shown in Table 5, for this set of experiments, the 27-group library always calculates supercritical, up to 4% above critical. The 238-group library calculations, shown in Table 6, vary by about 1.5% both above and below critical.

The third category consists of the ten thermal experiments listed in Table 4 having the Koponen citation numbers 111-**. They consist of highly enriched ^{233}U nitrate or fluoride solutions in either an aluminum cylinder or sphere surrounded by either a void or an hydrogenous reflector.[7] Experiments 111-01, -02, and -03 consist of highly enriched uranyl nitrate cylinders surrounded by paraffin reflectors. The calculations using the 27-group library are about 3% above critical for these cases, whereas the calculations using the 238-group library are critical within $\pm 3\sigma$. Experiment 111-20 is a highly enriched uranyl nitrate cylinder surrounded by a water reflector. The calculation using the 27-group library is less than 1% high, whereas the calculation using the 238-group library is about 1.5% low. This is the only case where the 27-group library calculated closer to critical than the 238-group library for this set of experiments. Experiments 111-21 and -22 are highly enriched uranyl fluoride-filled cylinders surrounded by paraffin. The calculations using the 27-group library are up to 5% above critical for these cases whereas the calculations using the 238-group library are up to 2.5% above critical. Experiments 111-23, -24, and -25 are highly enriched uranyl- fluoride-filled spheres surrounded by water. The calculations using the 27-group library exceed critical by as much as 4% for these cases, whereas the calculations using the 238-group library are within 2.5% of critical. The final experiment, 111-26, is a bare uranyl-fluoride-filled aluminum sphere. The calculation using the 27-group library is about 1.5% above critical, whereas the calculation using the 238-group library is critical within 2σ.

The energy of the average lethargy-causing fission (EALCF) is about 1 MeV for the 10 fast neutron experiments but only 0.1 eV for the thermal neutron experiments. No experiments containing ^{233}U fall within the range of 1.0 eV to 10 keV, where most of the resonance data resides. The fast neutron set of experiments and the thermal neutron set of experiments were chosen because they examine energies above and below this range of interest.

The set of 24 experiments contain all the materials and geometries used in the ACB, with the exception of carbon. The charcoal in the ACB is not significant from a neutronics standpoint (i.e., it has no significant reactivity effect). Its primary function is as an absorber of UF_4 and as a bulk medium that limits the amount of water in the ACB in case of flooding. The first 10 experiments, 1727-**, are all fast experiments. Since no epithermal experiments are available, a mixture of fast and thermal experiments was examined. The 1211-** and 111-** sets of experiments have most of the fission being produced by thermal neutrons. The array experiments, 1211-**, are type-304ss cylinders filled with uranyl nitrate. Since the ACB uses a 6-in. Sch 10 type-304ss pipe, experiments that use type-304ss were included. The final set of experiments, 111-**, are solution-filled cylinders and spheres. The solution cylinders represent the geometric configuration of the ACB; the spheres examine fluoride, which the ACB contains in the form of UF_4.

Table 4. Characteristics of ^{233}U critical benchmark experiments

Koponen citation ID	H/U atom ratio	Wt % ^{233}U	U density (g/cm^3)	Geometry	Material	Reflector
1727.01	0	98.11	18.42	Sphere	Metal	Bare
1727.02	0	98.11	18.42	Sphere	Metal	U-nat
1727.03	0	98.2	18.64	Sphere	Metal	U-nat
1727.04	0	98.2	18.62	Sphere	Metal	U-nat
1727.05	0	98.2	18.64	Sphere	Metal	W-alloy
1727.06	0	98.2	18.62	Sphere	Metal	W-alloy
1727.07	0	98.2	18.64	Sphere	Metal	Be
1727.08	0	98.2	18.62	Sphere	Metal	Be
1727.14	0	98.11	18.62	Sphere	Metal	U-235
1727.15	0	98.11	18.64	Sphere	Metal	U-235
1211.03	71.2	98.7	0.333	Cylinder, array	Nitrate	Polyethylene
1211.04	71.2	98.7	0.333	Cylinder, array	Nitrate	Polyethylene
1211.07	116.1	98.7	0.204	Cylinder, array	Nitrate	Polyethylene
1211.08	116.1	98.7	0.204	Cylinder, array	Nitrate	Polyethylene
111.01	56.8	98.7	0.386	Cylinder	Nitrate	Paraffin
111.02	66.1	98.7	0.340	Cylinder	Nitrate	Paraffin
111.03	143.1	98.7	0.169	Cylinder	Nitrate	Paraffin
111.20	399.7	98.7	0.063	Cylinder	Nitrate	Water
111.21	152.0	98.7	0.167	Cylinder	Fluoride	Paraffin
111.22	764.9	98.7	0.033	Cylinder	Fluoride	Paraffin
111.23	413.6	98.7	0.062	Sphere	Fluoride	Water
111.24	384.9	98.7	0.067	Sphere	Fluoride	Water
111.25	654.4	98.7	0.040	Sphere	Fluoride	Water
111.26	376.0	98.7	0.068	Sphere	Fluoride	Bare

Table 5. Validation results using the 27GROUPNDF4 cross-section library

Koponen citation ID	Energy (eV) of average-lethargy-causing fission	k_{eff} ($\pm\sigma$)
1727.01	9.568E+05	0.9636 (0.0016)
1727.02	8.967E+05	0.9803 (0.0015)
1727.03	9.149E+05	0.9766 (0.0016)
1727.04	9.217E+05	0.9711 (0.0017)
1727.05	7.671E+05	0.9737 (0.0016)
1727.06	8.400E+05	0.9684 (0.0016)
1727.07	5.771E+05	0.9857 (0.0017)
1727.08	7.602E+05	0.9801 (0.0017)
1727.14	9.338E+05	0.9767 (0.0015)
1727.15	9.308E+05	0.9791 (0.0016)
1211.03	2.956E-01	1.0391 (0.0020)
1211.04	1.997E-01	1.0386 (0.0021)
1211.07	1.423E-01	1.0277 (0.0021)
1211.08	1.435E-01	1.0144 (0.0021)
111.01	4.085E-01	1.0279 (0.0023)
111.02	3.155E-01	1.0276 (0.0022)
111.03	1.082E-01	1.0255 (0.0023)
111.20	4.495E-02	1.0073 (0.0018)
111.21	1.012E-01	1.0456 (0.0024)
111.22	3.253E-02	1.0120 (0.0017)
111.23	4.405E-02	1.0243 (0.0019)
111.24	4.613E-02	1.0392 (0.0019)
111.25	3.745E-02	1.0217 (0.0017)
111.26	5.464E-02	1.0111 (0.0020)

Table 6. Validation results using the 238GROUPNDF5 cross-section library

Koponen citation ID	Energy (ev) of average-lethargy-causing fission	k_{eff} ($\pm\sigma$)
1727.01	1.125E+06	0.9923 (0.0016)
1727.02	1.008E+06	0.9988 (0.0016)
1727.03	1.061E+06	1.0002 (0.0016)
1727.04	1.082E+06	0.9983 (0.0016)
1727.05	8.719E+05	1.0019 (0.0016)
1727.06	9.776E+05	1.0016 (0.0017)
1727.07	7.616E+05	1.0009 (0.0016)
1727.08	9.451E+05	0.9944 (0.0016)
1727.14	1.094E+06	0.9953 (0.0017)
1727.15	1.066E+06	1.0031 (0.0016)
1211.03	3.324E-01	1.0136 (0.0019)
1211.04	3.457E-01	1.0134 (0.0019)
1211.07	1.718E-01	1.0016 (0.0024)
1211.08	1.726E-01	0.9887 (0.0021)
111.01	4.637E-01	0.9972 (0.0022)
111.02	3.652E-01	1.0010 (0.0021)
111.03	1.327E-01	1.0066 (0.0022)
111.20	6.009E-02	0.9875 (0.0018)
111.21	1.254E-01	1.0207 (0.0021)
111.22	4.520E-02	1.0057 (0.0017)
111.23	5.916E-02	1.0083 (0.0020)
111.24	6.165E-02	1.0226 (0.0019)
111.25	4.780E-02	1.0095 (0.0017)
111.26	7.214E-02	0.9963 (0.0020)

5. VALIDATED RESULTS AND CONCLUSIONS

The analysis of the set of 24 experiments is considered an acceptable validation for the ACB. The entire analysis was done using the SCALE-4.3 criticality safety code package.[8] All geometries and materials, with the exception of carbon, are present in the set of experiments. All experiments were critical. Using the 238-group library, k_{eff} ranges from 1.5% below critical to 2.5% above critical. Using the 27-group library, k_{eff} ranges from 4% below critical to 5% above critical. Since calculations using the 238-group library reproduce the experimental results more closely, this library is used to do the final analysis for the ACB.

Table 7 contains the results of four conditions that represent the moderation extremes for the ACB. The difference between the cases in Table 7 and those contained in Tables 1, 2, and 3 is the cross-section library used. The cases in Table 7 use the 238-group ENDF/B-V cross-section library whereas the cases in Tables 1, 2, and 3 use the 27-group ENDF/B-IV cross-section library. The k_{eff} of the same problem calculated with the two libraries can vary by up to 5%. All cases in Table 7 assume the uranium, in the form of UF_4 containing 2.0 kg of ^{233}U, has the same isotopic distribution as the uranium in the FDTs. The UF_4 is uniformly mixed in the top 12 in. of charcoal of the ACB. The first case assumes that no water is present in or around the ACB. This case, which represents the current state of the ACB, has a k_{eff} less than 0.1; thus there is no criticality safety concern. The EALCF is 1.4 keV, thus putting this case in the intermediate energy range. The carbon does moderate the neutrons some, but not enough to slow them down below the resonance region.

The second case assumes that no water is inside the ACB (no internal flooding), and a 3-ft water reflector surrounds it. This case is also well undermoderated and has an EALCF of 1.6 eV. The thermal neutrons returning from the water reflector dominate the reactivity. This case is also well subcritical.

The third case assumes water internal to the ACB but no reflector. This case is also well undermoderated, having an EALCF of 1.55 eV. The water slows the neutrons into the epithermal range before they fission or escape. This case is also well subcritical.

The fourth case assumes that the ACB is both moderated and surrounded by a 3-ft-thick water reflector. The EALCF for this case, 0.6 eV, is about 25% higher than that for the highest thermal experiment examined. The case is well subcritical even after adding in a negative bias of 1.5%, the largest negative bias encountered when using the 238-group library. The k_{eff} for this case is 3.5% lower than the k_{eff} for the same case shown in Table 1 that uses the 27-group library. Unfortunately, the amount and distribution of the uranium are not known well enough to establish this case as the possible configuration closest to critical. Nevertheless, given the uncertainties associated with the amount and distribution of uranium, the analysis clearly shows that criticality safety is only a significant concern if water is present in and around the ACB.

Table 7. Results of MSRE ACB validation cases using the 238GROUPNDF5 cross-section library

Case	Energy (ev) of average lethargy causing fission	k_{eff} ($\pm\sigma$)
Dry, unreflected	1.397E+03	0.0792 (0.0005)
Dry, reflected	1.611E+00	0.3764 (0.0013)
Flooded, unreflected	1.554E+00	0.6248 (0.0020)
Flooded, reflected	6.077E-01	0.9073 (0.0022)

6. REFERENCES

1. D. F. Hollenbach and C. M. Hopper, *Criticality Safety Study of the MSRE Fuel Drain Tank Cell in Building 7503*, ORNL/TM-12646, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., January 1994.

2. Personal Correspondence, D. W. Ramey to B. D. Patton, *Technique for Estimating the Uranium Content in the MSRE Auxiliary Charcoal Bed (ACB)*, September 12, 1994.

3. R. T. Primm III, *Criticality Safety Studies of Building 3019 Cell 4 and In-Line Storage Wells*, ORNL/TM-12374, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., November 1993.

4. B. L. Koponen et al., *Nuclear Criticality Experiments from 1943 to 1978, An Annotated Bibliography*, UCRL-52769, University of California, Livermore, April 1979.

5. H. C. Paxton, *Los Alamos Critical-Mass Data*, LA-3067-MS, Los Alamos Natl. Lab., December 1975.

6. J. T. Thomas, *Critical Experiments with Aqueous Solutions of Uranium-233* $O_2(NO_3)_2$, ORNL-4280, Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., 1968.

7. J. K. Fox, L. W. Gilley and E. R. Rohrer, *Critical Mass Studies, Part VIII, Aqueous Solutions of* ^{233}U, ORNL-2143, Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., September 1959.

8. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Vols. I-III, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4) (April 1995). Available from Radiation Shielding Information Center as CCC-545.

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