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Reactor Waste and Associated Fuel Hardware for  
Decommissioning of EBR-II

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# Alternate Form and Placement of Short Lived Reactor Waste and Associated Fuel Hardware for Decommissioning of EBR-II

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## ABSTRACT

This is the 100 word abstract ??? (write after the conclusion is done.

## INTRODUCTION

Upon the termination of EBR-II operation in 1994, the mission has progressed to decommissioning and waste cleanup of the facility. The simplest method to achieve this goal is to bury the raw fuel and activated steel in an approved burial ground or deep geologic repository. While this might be simple, it could be very expensive, consume much needed burial space for other materials, and leave large amounts of fissile material easily available to future generations. Also, as with any operation, an associated risk to personnel and the public from the buried waste exists. To try and reduce these costs and risks, alternatives to burial are sought. One alternative explored here for EBR-II is to condition the fuel and store the fission products and steel either permanently or temporarily in the sealed primary boundary of the decommissioned reactor.

To achieve this goal, various tasks must be performed. Ultimately, a safety analysis must determine if the project is safe enough to be implemented; however, basic questions must be answered to perform an analysis. The first problem is to identify which sub-assemblies are going to be conditioned and their current composition and decay time. The next problem is to identify the conditioning process and determine the composition and form

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Table 1: Inventory of EBR-II Sub-assembly Pins and Heavy Metal Mass

Sub-assembly Type	Description	Pins	kgHM
Mark-IIIC/IICS	Control Rods	3807	150
Mark-IIII/IIIA	Drivers	9709	680
Ternary	Drivers	3	0.03
Mark-IIIC/IICS	Experiments	135	4
Mark-IIII/IIIA	Experiments	1813	123
Ternary	Experiments	469	29
Misc.	Experiments	637	62
Blanket in Rx	Depleted Urn	5928	14,455
Blanket in HFEF	Depleted Urn	500	1,251
Blanket in RSWF	Waste	7486	6,464

of the waste streams. The volume, mass, heat, and curie load of the waste streams needs to be determined so a waste-assembly can be designed. The reactor vessel and internals need to be analyzed to determine if they can handle these loads. If permanent storage is the goal, then mechanisms for placing the waste-assembly in the reactor vessel and sealing the vessel are needed. If temporary storage is the goal, then mechanisms for waste-assembly placement and retrieval are needed.

This paper answers the technical questions of volume, mass, heat, and curie loads while just addressing the other questions found in a safety analysis. The final conclusion will compare estimated risks from the burial option and this option.

## INVENTORY AND COMPOSITION

If anything is to be done with the spent fuel from EBR-II, it must be identified and a composition determined. For a commercial nuclear power facility, this is a fairly easy task. For EBR-II personnel, this task is complicated by the fact that a reprocessing campaign was pursued in the 1960's, some of the sub-assemblies were taken apart for destructive and non-destructive testing, and some of the fuel pins existed in more than one sub-assembly. Fortunately, the amount of fuel in this category is small. The majority of the fuel and blanket is in the reactor vessel, the Hot Fuel Examination Facility (HFEF), and the Radioactive Scrape and Waste Facility (RSWF). Additionally, 6.5 metric tonnes of heavy metal was declared to be waste by the United States Department of Energy (DOE) and exists in RSWF. Approximately 2 metric tonnes of heavy metal from Mark-II fuel is stored at the Idaho Chemical Processing Plant (ICPP) and 0.1 metric tonnes of heavy metal from Mark-II fuel is stored on site. Table 1 shows the inventory identified for conditioning which consists of 1.048 metric tonnes of heavy metal in driver, control, and miscellaneous sub-assemblies and 15.706 metric tonnes in blanket sub-assemblies, and 6.464 metric tonnes declared as waste.

Table 2: Constituents of the Elemental Groups

Group Name	Group	Elements or materials in the group
Uranium	Urn	U
Higher Actinides	TRU	Pu Ac Th Pa Np Am Cm
Active Metal FP	AM	Na Rb Cs Fr Be Mg Ca Sr Ba Ra Sc Sm Eu Gd Tb Li K F Cl Br I At
Volatile FP	VFP	He Ne Ar Kr Xe Rn
Rare Earth FP	RE	Y La Ce Pr Nd Pm Dy Ho Er
Noble Metal FP	NM	Zr Cd B Al Ti V Cr Mn Fe Co Ni Cu Zn Ga Ge As Sc Nb Mo Tc Ra Rh Pd Ag In Sn Sb Te Ta W Tl Pb Bi Po
Other	OTH	H C N O Si P S Tm Yb Lu Hf Re Os Ir Pt Au Hg

Table 3: Elemental Group Densities Over Time in Grams per cm<sup>3</sup> of Fresh Mark-III A Fuel

Element	Charge	Discharge	10min	1yr	10yr	100yr	1,000yr
Urn	3.98E+00	3.58E+00	3.58E+00	3.58E+00	3.58E+00	3.58E+00	3.58E+00
TRU	0.00E+00	2.11E-02	2.12E-02	2.12E-02	2.12E-02	2.11E-02	2.06E-02
AM	4.12E-01	4.97E-01	4.96E-01	4.97E-01	4.98E-01	4.93E-01	4.92E-01
VPS	0.00E+00	5.08E-02	5.08E-02	5.08E-02	5.06E-02	5.04E-02	5.04E-02
RE	0.00E+00	1.11E-01	1.10E-01	1.10E-01	1.07E-01	1.07E-01	1.07E-01
NM	2.16E+00	2.29E+00	2.29E+00	2.29E+00	2.29E+00	2.30E+00	2.30E+00
OTH	8.58E-03	8.58E-03	8.58E-03	8.58E-03	8.58E-03	8.58E-03	8.58E-03
Tot	6.56E+00	6.56E+00	6.56E+00	6.56E+00	6.56E+00	6.56E+00	6.56E+00

In order to keep the reportable data on compositions in a manageable form, elemental groups have been created that correspond to real elemental separations occurring in the conditioning process. Table 2 shows the breakdown.

The complete elemental composition of each sub-assembly is, of course, unknown. An average number for each fuel type can be calculated based on burnup and reactor power with the ORIGEN code. Table 3 shows the element group densities of Mark-III A fuel as a function of decay time and Table 4 shows the decay heat associated with this fuel. The burnup used for this initially fresh fuel to achieve conservative results is 10 atom percent at a reactor power of 62.5 MW which gives a reactor residence time of 433 days. The total energy release for EBR-II is  $9 \times 10^6$  MWhrs.

## CONDITIONING PROCESS

There are many activities involved in conditioning the fuel from EBR-II: reactor unloading, sub-assembly wash, transfer to HFEB, blanket assembly reduction, waste can loading, transfer to the RSWF, retrieval from RSWF, waste can unloading, transfer to Fuel Cycle Facility (FCF), electrochemical treatment, transfer of interim storage prod-

Table 4: Elemental Group Decay Heat Over Time in Watts per cm<sup>3</sup> of Fresh Mark-IIIA Fuel

Element	Charge	Discharge	10min	1yr	10yr	100yr	1,000yr
Urn	5.05E-06	9.25E-02	4.77E-06	4.77E-06	4.78E-06	4.77E-06	4.76E-06
TRU	0.00E+00	5.19E-02	7.14E-05	7.11E-05	6.88E-05	5.29E-05	3.62E-05
AM	0.00E+00	1.75E+01	8.67E-03	7.77E-03	5.48E-03	6.62E-04	1.94E-08
VPS	0.00E+00	4.22E+00	2.09E-04	1.96E-04	1.09E-04	3.36E-07	1.24E-08
RE	0.00E+00	1.07E+01	7.61E-02	3.47E-02	4.75E-03	5.07E-04	2.02E-12
NM	0.00E+00	1.27E+01	1.84E-02	5.88E-03	4.43E-05	3.91E-07	3.38E-07
OTH	0.00E+00	6.17E-25	6.17E-25	6.17E-25	6.17E-25	6.10E-25	5.47E-25
Tot	5.05E-06	4.52E+01	1.03E-01	4.86E-02	1.05E-02	1.23E-03	4.14E-05

ucts and waste form samples, permitting and other activities. This paper will just deal with the electrochemical waste streams and the mechanical formation of the final waste products.

Four types of waste streams are generated with this process. One type does not concern this paper: volatile fission products (VFP). These are released when the fuel pin cladding is pierced in the chopping phase and are cryogenically recovered from the argon gas atmosphere used in FCF. They can then be released to the atmosphere or used for various tasks in industry like krypton self-powered emergency egress lights or xenon tag gases for nuclear medicine breathing and other studies. The other types of waste streams must be processed into a form that reduces their risk of escape and contamination. The noble metal fission products (NM) are recovered from the cadmium pool and as cladding hulls. The active metal fission products (AM) are recovered from the molten salt electrolyte. The rare earth fission products (RE) are recovered from the higher actinide product and the molten salt phase. The two product streams of uranium and TRU ingots are also not discussed in this paper.

## WASTE FORMS

The waste stream treatment problem does not have one solution. The AM's and RE's exist in a different part of the electro-refiner than the NM's. Different methods are needed to recover them. The contaminated molten salt will be passed through a zeolite column to separate the AM's and RE's from the molten salt with the salt passing through the column and recycled back into the electro-refiner. Once the zeolite is loaded with the fission products, a hot isostatic press will be used to create a final waste form. The NM's (including cladding hulls) will be cast in an alloy mixture for a final waste form. To extract the RE's from the TRU's, a pyrocontactor will be used. With these waste forms, a final physical configuration of these materials can be designed to allow the storage of the wastes on the reactor grid in waste-assemblies.

## WASTE-ASSEMBLY DESIGN

With the physical and design information mentioned above, an analysis can be performed to design the waste-assemblies and their characteristic behavior in the reactor vessel. The ORIGEN code is used to generate an average composition density along with heat and curie loads. From this data, the environment in the reactor vessel for the 300 to 500 year storage time can be predicted. A safety analysis would then predict the risks associated with this storage plan.

This analysis is based on some assumptions which are that the bond sodium goes with the AM's, the cladding hulls are shown with the NM's, the pole pieces are not included with the NM's, all the RE's are in the salt phase, and the fuel has cooled for an average of 10 years before being put into the reactor vessel. Therefore, the discharge column from Table 3 and the 10 year column from Table 4 are used as a starting point. Also, an initial heavy metal loading of  $3.98 \frac{\text{gm}}{\text{cm}^3}$  is assumed.

The physical configuration of the waste-assembly will be the same as a 61 pin Mark-III driver except that the plenum and shield regions are filled with waste. This is used as a basis design to allow a large air flow to pin aspect ratio for better heat transfer characteristics. The waste-assembly will be the standard hex can configuration so that no modifications will need to be made to the existing equipment in the reactor vessel and 637 waste-assemblies can exist on the reactor grid plate.

Estimates for the physical parameters of the waste forms must be made to establish an analysis path. The only parameters needed are the densities of the waste forms and the loading of the waste matrix. The density and loading of the zeolite glass is estimated at  $2.5 \frac{\text{gm}}{\text{cm}^3}$  and 10%. For the stainless steel alloy, the estimates are  $7 \frac{\text{gm}}{\text{cm}^3}$  and 20%. From these parameters, various quantities can be calculated.

If Mark-III fuel was used over EBR-II's history and burned at the specifications above, 395 kg of fission products and steel waste would be generated (minus pole pieces). Of the total, there would be 68 kg of AM's, 15 kg of RE's, and 312 kg of NM's. This would create 911 kg of loaded zeolite and 1,871 kg of alloyed stainless steel or a total of 513 waste-assemblies: 242 AM's, 54 RE's, and 217 NM's.

If the fuel had an average cooling time of 10 years, then the total heat load would be 1.4 kW when the waste-assemblies were placed in the reactor vessel. The heat load per waste-assembly would be 3 watts for an AM waste-assembly, 12 watts for an RE, and 0.03 watts for an NM. The heat load would of course decrease over time. With this magnitude of heat load, no forced convection cooling is envisioned; however, this will need to be analyzed in detail for the safety analysis.

The radiological consequences of this storage plan are still under study. About half of the decay heat after 10 years is from gamma heating. Table 5 shows the curie content of the elemental groups. The majority of the radioactive decays reside in the active metals and the rare earths. The energy spectrum of the gammas generated are being analyzed and those results will be used to determine the radiation environment of the reactor vessel.

Table 5: Elemental Group Curie Content Over Time in Curies per cm<sup>3</sup> of Fresh Mark-III A Fuel

Element	Charge	Discharge	10min	1yr	10yr	100yr	1,000yr
Urn	1.76E-04	4.34E+01	1.66E-04	1.66E-04	1.66E-04	1.66E-04	1.66E-04
TRU	0.00E+00	3.76E+01	2.41E-03	2.39E-03	2.26E-03	1.68E-03	1.17E-03
AM	0.00E+00	1.13E+03	3.67E+00	3.46E+00	2.69E+00	3.31E-01	3.12E-05
VPS	0.00E+00	3.67E+02	1.39E-01	1.30E-01	7.30E-02	2.18E-04	3.55E-07
RE	0.00E+00	9.28E+02	2.06E+01	1.00E+01	1.05E+00	9.19E-02	6.81E-10
NM	0.00E+00	1.14E+03	3.81E+00	1.24E+00	1.37E-02	6.59E-04	3.10E-04
OTH	0.00E+00	2.08E-21	2.08E-21	2.08E-21	2.08E-21	2.06E-21	1.85E-21
Tot	1.76E-04	3.65E+03	2.82E+01	1.48E+01	3.83E+00	4.25E-01	1.68E-03

## MECHANICAL CONSIDERATIONS

Once the heat and curie loads are determined, other concerns need to be addressed. The primary concern is the grid plate and its support structure where the waste-assemblies will reside; however, a critical loading experiment was performed before the liquid sodium was in place. Therefore, an unirradiated grid plate can hold and allow manipulation of the waste-assemblies without the static sodium load. A materials analysis will determine if an irradiated grid plate will function the same over the time period expected. Another concern is the functioning of the grid plate in the new radiation environment for the expected time period. The safety analysis will be used to determine this factor.

Other concerns exist. When the liquid sodium is drained from the vessel, residual sodium is present. At this moment, the residual sodium is not deemed a risk because as the sodium freezes, its reactivity rate with water goes down. Oxide crusts will eventually form over any residual pools and render them non-reactive. Upon detailed analysis, if a noble gas cover is needed to minimize the sodium water reaction, then the concern of gas leakage will need to be addressed. A gas cover complicates the sealing of the reactor vessel and an analysis of consequences from gas leakage in or out will be needed. Whatever atmosphere is in the reactor vessel, it is the only heat transfer medium available to move the heat from the reactor components and the waste-assemblies to the tank sides for final transfer to the environment. A thermal hydraulic analysis will be needed to determine if the reactor's atmosphere can handle the heat transfer. However, from the analysis above, all the waste-assemblies are generating 1.4 kW of heat. Not enough heat is generated to melt the waste-assemblies or to need forced convection. The list of concerns continues; however, a complete safety analysis will cover these type of questions.

## CONCLUSION

With this initial physical analysis of waste placement back into the reactor vessel, it appears to warrant further engineering and regulator analysis. On a very conservative estimate of the volume of waste to be produced, it will fit back into the reactor vessel where it was generated in a manner which will entail very little mechanical redesign

of systems. Engineering and regulatory concerns still must be addressed and problems solved; but, continued analysis is recommended.

Of greatest concern is the relative risk of this storage plan over the deep geological repository which currently has political problems not reflected in its risk assessment. In the storage plan advocated in this paper, no fissile material is present so criticality and proliferation concerns do not exist. In the EBR-II probability risk assessment (PRA), the major concern was from a large earthquake causing the tank support to fail and the reactor to fall away from the control rod drives and inserting reactivity into the core as it falls. This is not a concern with this plan. The major concern is will the material hold for the 300 to 500 years necessary to render the waste less radioactive than the original ore from whence it came. From the EBR-II PRA, the risk is ????. From best estimates, the Yucca Mountain repository risk is ????. We feel that a ?100? fold decrease in the EBR-II risk would be seen if it was converted to a waste repository and therefore be a safer repository than Yucca Mountain.