HNF-EDC-05 - 27696

Page 1 of 2

EDC (ENGINEERING DOCUMENT CHANGE) FORM

Document Identification

1. Change Title:

Change pin container listing in Table 1 of FFTF-17294 to account for separating SRF-3 and SRF-4 and transfer of inner fuel bundles into ID-69E's.

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Additional Reviewers:

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11. Change Description (description and reason for requested change):

Remove Applied Technology designation per DOE-NE (Reference DOE-RL: 05-AMRC-0363).

Add ID-69E-60 and ID-69E-61 to the Table 1 listing. These ID-69's contain inner fuel bundles from the SRF-3 and SRF-4 assemblies. These inner fuel bundles are acceptable for loading into Interim Storage Casks.

Approvals

12. Change Originator

JR Thielges

Print/Signature/Date

T/DA

SA Chastain

Print/Signature/Date

Engineering Management/TA Manager

SW Hiller

Print/Signature/Date

Title Nuclear Safety

SU Zaman

Print/Signature/Date

Title Quality Assurance

ME Riste

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Title SNF Interface

RL McCormack

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13. Document Index

Action	Number	Title	Rev (being issued)	Change Page(s)	Config Baseline
DR	FFTF-17294	Review of Fast Flux Test Facility (FFTF) Fuel Experiments for Storage in Interim Storage Casks (ISCs)	1	All	☐

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Revision 1

Review of Fast Flux Test Facility (FFTF) Fuel Experiments for Storage in Interim Storage Casks (ISCs)

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
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Fluor Hanford
P.O. Box 1000
Richland, Washington

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Revision 1
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REVIEW OF FAST FLUX TEST FACILITY (FFTF) FUEL EXPERIMENTS FOR STORAGE IN INTERIM STORAGE CASKS (ISCs)

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REVIEW OF FAST FLUX TEST FACILITY (FFTF) FUEL EXPERIMENTS FOR STORAGE IN INTERIM STORAGE CASKS (ISCs)

1.0 INTRODUCTION

Appendix H, Section H.3.3.10.11 of the Final Safety Analysis Report (FSAR), provides the limits to be observed for fueled components authorized for storage in the Fast Flux Test Facility (FFTF) spent fuel storage system. Currently, the authorization basis allows standard driver fuel assemblies (DFA), as described in the FSAR Chapter 17, Section 17.5.3.1, to be stored provided decay power per assembly is ≤ 250 watts, post-irradiation time is four years minimum, average assembly burn-up is 150,000 MWD/MTHM maximum and the pre-irradiation enrichment is 29.3% maximum (per H.3.3.10.11). In addition, driver evaluation (DE), core characterizer assemblies (CCA), and run-to-cladding-breach (RTCB) assemblies are included based on their similarities to a standard DFA. Ident-69 pin containers with fuel pins from these DFAs can also be stored. Section H.3.3.10.11 states that fuel types outside the specification criteria above will be addressed on a case-by-case basis. There are many different types of fuel and blanket experiments that were irradiated in the FFTF which now require offload to the spent fuel storage system.

Two reviews were completed for a portion of these special type fuel components to determine if placement into the Core Component Container (CCC)/Interim Storage Cask (ISC) would require any special considerations or changes to the authorization basis. Project mission priorities coupled with availability of resources and analysts prevented these evaluations from being completed as a single effort. Areas of review have included radiological accident release consequences, radiological shielding adequacy, criticality safety, thermal limits, confinement, and stress. The results of these reviews are available in WHC-SD-FF-RPT-005, Rev. 0 and 1, *Review of FFTF Fuel Experiments for Storage at ISA*, (Reference 1), which subsequently allowed a large portion of these components to be included in the authorization basis (Table H.3.3-21). The report also identified additional components and actions in Section 3.0 and Table 3 that require further evaluation. The purpose of this report is to evaluate another portion of the remaining inventory (i.e., delayed neutron signal fuel, blanket assemblies, highly enriched assemblies, newly loaded Ident-69 pin containers, and returned fuel) to ensure it can be safely offloaded to the FFTF spent fuel storage system.

2.0 EVALUATION SUMMARY

Table 1 provides a listing of the fuel, including both intact assemblies and pin containers, included in the scope of this report. The list includes components identified in Table 3 of Reference 1, and also components that were returned to the FFTF from the Idaho National Environmental Engineering Laboratory (INEEL), Argonne National Laboratory-West (ANL-W) and the 300 Area after Reference 1 was issued. The ensuing sections provide a summary of the

evaluation results, which have verified that the existing safety analysis bounds these components as well. Therefore, the Table 1 components can be added to Table H.3.3-21 of the FSAR without any special considerations or changes to the authorization basis.

2.1 Returned Fuel

The fuel that was returned from ANL-W and the 300 Area is stored in 69C #6, 11PB40, 69C #34, and 69D #50. Other pins originating from the same assemblies as the returned pins are stored in significantly larger quantities in other pin containers that have been previously approved for storage in the CCC/ISC storage system. Therefore, these returned pins are bounded by the previous assessments provided in Reference 1, except for 69C #34, which contains encapsulated pins. These types of pins are addressed below in Section 2.7, and are also found to be acceptable.

2.2 Radiological Accident Releases

Reference 1 concluded that all of the oxide, carbide, nitride, and blanket assemblies were within the bounds of the existing analyses. In addition, storing fuel assemblies and pin containers in the same CCC/ISC was determined to be acceptable. Therefore, all the fuel listed in Table 1 is acceptable in terms of the radiological accident analyses.

2.3 Radiological Shielding

Reference 1 concluded that the ISC shielding is adequate for the inventory of experimental fuel with the exception of the sodium-bonded metal fuel, carbide fuel, and blanket assemblies. Further shielding evaluations were completed in HNF-SD-FF-CN-010, Rev. 0 (Reference 2) to evaluate the sodium-bonded metal, carbide, and blanket fuel inventory. Irradiation parameters and history were obtained from the FFTF data base maintained for all FFTF assemblies. The gamma ray spectra were reviewed to determine whether they were within the envelope of gamma rays from a standard DFA with 150 Mwd/kg burnup. The conclusion of the study was that only one assembly (MFF-3, 114 Mwd/kg BU) has any gamma rays outside the standard DFA source envelope. The gamma rays in the 2.25 MeV energy group are about 40% higher than the standard DFA. However, the dose rate through the side of the ISC will be about 20% lower than the standard DFA because most of the dose rate is from higher intensity and lower energy gamma rays.

Attachment 1 contains an evaluation of the experiments and repackaged fuel returned to the FFTF from the ANL-W. This evaluation concluded that the radioactive inventory of the returned fuel would be less than the inventory of a typical worst case driver fuel assembly. Therefore, all of the fuel in Table 1 is acceptable in terms of radiological shielding.

2.4 Criticality

The fuel specified in Table 1 is permitted to be loaded into CCCs for storage in ISCs based on criticality considerations addressed in HNF-SD-CSA-00003, *Classification of FFTF Fuel Assemblies and Fuel Pins*, Revisions 0 thru 1B, and is documented in the FSAR, Chapter 20. Analysis has shown that under all normal and credible accident conditions, the listed assemblies will remain subcritical.

2.5 Thermal

Attachment 2 provides an evaluation of four fuel types, including 1) blanket assemblies, 2) carbide fuel, 3) sodium bonded metal fuel, and 4) highly enriched fuel, to determine if the stress rupture behavior of these assemblies would be the same as that previously analyzed in Reference 3. The analysis in Reference 3 was used in developing a maximum initial fuel pin temperature limit of 900°F for ISC storage. As indicated in Attachment 2, the blanket assemblies and the highly enriched fuel assemblies, SRF-3 and SRF-4, are covered by the ISC design fuel pin temperature limit of 900°F. Data used to support this assessment is provided in Attachment 4.

A thermal evaluation of all the experimental fuel assemblies and pin containers with discrete fuel pins from experimental assemblies that differed geometrically from a standard DFA is provided in Reference 4. Different fuel types were also evaluated. Based on the summary presented in Reference 1, all of the experimental fuel assemblies will have peak cladding temperatures lower than a standard DFA when stored in an ISC under identical conditions, except for the multi-duct fuel test assemblies SRF-3 and SRF-4. If these assemblies are limited to approximately 25% of the power (62.5 watts) that is allowed for a standard DFA, then the peak pin temperatures will be equal. As of March 6, 1996, SRF-3 was 53.50 watts and SRF-4 was 57.31 watts, below the 62.5 watt limit. Today (nine years later), the decay heat of these two assemblies is even lower. Therefore, the pin containers and intact assemblies listed in Table 1 are acceptable based on the design fuel pin temperature limit and thermal evaluations.

2.6 Stress

Reference 1 provided a summary of an evaluation based on the worst case loading of the FFTF experimental fuel inventory which demonstrated that maximum container pressures would be less than the value used for the CCC/ISC design basis calculations and hydrostatic test pressure. Intermixing of pin containers and intact fuel assemblies was also found to be acceptable. The only question noted was that if less than a full load (< six fueled components) is in the CCC, the ISC impact limiter will need to be evaluated. For information on full CCC payloads, refer to Table A3-1, *DFA and Pin Container Payload Summaries (Fueled)* (Reference 5). Therefore, the pin containers and intact assemblies listed in Table 1 are acceptable based on the stress evaluations.

2.7 Delayed Neutron Signal Breached Fuel Pins in Long Term Storage

An evaluation of the planned method to clean and prepare breached fuel pins for long term storage in the CCC/ISC storage system is provided in Attachment 3. Fuel assemblies with known breached pins, i.e., those that exhibited a delayed neutron (DN) signal when in the reactor, are not put through the conventional wash cycle in the Sodium Removal System because of criticality safety concerns documented in Section 20.4.2.D.1.4 of the FSAR. Current planning is to disassemble the DN signal assemblies, identify and individually wipe down each breached pin, encapsulate them in an outer protective stainless steel tube, and then store them in an "Ident-69" or similar container for storage in an ISC. Fewer than five fuel assemblies contain known or suspected DN leakers. Pin containers 69C #34 and 69C #27 contain encapsulated breached pins (3 pins and 1 pin, respectively).

Attachment 3 indicates that the low probability of any significant amount of sodium inside breached pins (if sodium is present), the ability to detect any such sodium if it were present by identifying and weighing the breached pin, added to the planned encapsulation of breached pins, the inert atmosphere inside the ISC/CCC, and the capability to obtain samples of the cask atmosphere prior to opening, provide sufficient conservatism and defense-in-depth to conclude that no additional measures are required to mitigate the potential for sodium reactions inside the ISC/CCC.

3.0 TEST FUEL INVENTORY REQUIRING FURTHER EVALUATION

Only two categories of fuel, sodium bonded metal fuel and carbide fuel, require further evaluation. The required follow-on actions are described below and will need to be addressed in a subsequent revision to the FSAR.

3.1 Sodium Bonded Metal Fuel: MFF-1, MFF-2, MFF-3, MFF-4, MFF-5, MFF-6, IFR-1 (Ident 69 #4)

Current planning is to transfer this fuel to the INEEL for consolidated storage. If the INEEL, ANL-W facility is not readily available, this fuel may be loaded in the CCC/ISC storage system. If this option is pursued in the future, in addition to the analyses that have been completed to date for this fuel (described below), two additional actions would be required: 1) modify the SARP to allow future transport of this fuel in the ISC to the Canister Storage Building, and 2) to evaluate providing continuous monitoring of the ISC seal during the storage period (to support revision of the FSAR). It is also noted that the sodium bonded metal pins from MFF-1 and the sodium bonded carbide fuel pins in ACN-1 will be dispositioned with the metal fuel.

Completed Analyses:

A radiological accident release assessment was completed (Attachment 5). The conclusion of this evaluation is that the storage of the metal fuel assemblies does not result in the dose calculations exceeding existing ISC radiological analyses, even for beyond design basis accidents.

Reference 2 included a shielding evaluation of the metal fuel with conclusions consistent with those discussed above in Section 2.3.

Attachment 2 provided an evaluation of the fuel pin temperature limit and concluded that it is covered by the ISC design fuel pin temperature limit of 900°F. In addition, a thermal evaluation was completed for this inventory in Reference 4 and was found to be acceptable.

3.2 Carbide Fuel: FC-1, AC-3, ACN-1

Current planning is to place all the carbide fuel, except the sodium bonded fuel from ACN-1, in one CCC; therefore, all of this type of fuel will be addressed in a single offload campaign. An issue was raised by the British related to washing/handling of carbide fuel due to a concern that the fuel is pyrophoric. This issue was satisfactorily addressed in Reference 6 for washing the carbide fuel. An additional concern has been raised related to the long-term storage of carbide fuel due to its pyrophoric nature. This issue needs to be addressed prior to storing these assemblies in an ISC.

Completed Analyses:

A radiological accident release assessment was completed (Attachment 5). The conclusion of this evaluation is that the storage of the carbide fuel assemblies does not result in the dose calculations exceeding existing ISC radiological analyses, even for beyond design basis accidents.

Reference 2 included a shielding evaluation of the carbide fuel with conclusions consistent with those discussed above in Section 2.3.

Attachment 2 provides an evaluation of the fuel pin temperature limit which indicates the carbide fuel is covered by the ISC design fuel pin temperature limit of 900°F. In addition, a thermal evaluation was completed for this inventory in Reference 4 and was found to be acceptable.

5.0 CONCLUSIONS

Based on the aforementioned evaluations, the fuel listed in Table 1 can be added to Table H.3.3-21 of the FSAR without any special considerations or changes to the authorization basis.

FFTF-17294

Revision 1

6.0 REFERENCES

1. WHC-SD-FF-RPT-005, Rev. 1, *Review of FFTF Fuel Experiments for Storage at ISA*, B&W Hanford Co., Richland, WA, October 18, 1996.
2. HNF-SD-FF-CN-010, Rev. 0, *Shielding Evaluation of Special FFTF Fuel Assemblies in ISC*, Fluor Daniel Northwest, Richland, WA, April 11, 1997.
3. Letter, B. C. Gneiting to L. R. Besel, *Pin Temperature Limits to Restrict Stress Rupture Breaches During Long Term Storage of FFTF Fuel Assemblies*, January 27, 1992.
4. WHC-SD-FF-ER-100, Rev. 2, *Thermal Analysis of the Core Component Container Within the Interim Storage Cask*, Westinghouse Hanford Co, Richland, WA, October 8, 1996.
5. HNF-SD-TP-SARP-010, Rev. 1A, *Safety Analysis Report for Packaging (Onsite) Interim Storage Casks*, June 16, 2003.
6. FFTF-21711, Rev. 0, *Fast Flux Test Facility Carbide Fuel Washing Safety Review*, Fluor Hanford, Richland, WA, September 2004.

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Revision 1

Table 1. Evaluated Test Assemblies and Pin Containers			
Test Assembly or Pin Container	Serial Number	Fuel Category	Description
69C #27	DFA-16392 2144 FEX 2170 FEX	DNM Signal	Contains 1 failed encapsulated pin from a standard DFA which failed during reactor operation as well as: 12 pins from ACO-3 86 pins from MFA-1
SRF-3	2075 FEX	Highly Enriched	Multi-duct, highly enriched uranium test pins shielded by surrounding boron carbide pins.
SRF-4	2076 FEX	Highly Enriched	Multi-duct, highly enriched uranium test pins shielded by surrounding boron carbide pins.
ABA-1	2160 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
ABA-2	2161 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
ABA-3	2162 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
ABA-4	2163 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
ABA-5	2164 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
ABA-6	2165 FEX	Blanket Assembly	Core demonstration experiment (CDE) long-lifetime inner blanket used to simulate heterogenous core load. Contains 91 pins with HT9 cladding.
WBA-40	4183 FEX (WB-018)	Blanket Assembly	Radial blanket assembly for a reference 316 stainless steel LMR core system. Consists of 61 pins and 316 SS cladding.
WBA-41	4184 FEX (WB-042)	Blanket Assembly	Radial blanket assembly for a reference 316 stainless steel LMR core system. Consists of 61 pins and 316 SS cladding.
29 69C	2172 FEX (MBA-1)	Blanket Assembly	MBA-1 was a performance test of Japanese blanket assembly with 316 SS cladding. Contains 51 pins of depleted uranium oxide.
UO-1	2116 FEX	Blanket Assembly	Fuel test to confirm the performance of a future uranium dioxide fueled core (Series IV). Four types of fuel pins.

FFTF-17294

Revision 1

Table 1. Evaluated Test Assemblies and Pin Containers			
Test Assembly or Pin Container	Serial Number	Fuel Category	Description
AB-1	2114 FEX	Blanket Assembly/Gas Leaker	Characterized the behavior of a mixed oxide axially blanketed fuel and cladding system under prototypical operating conditions. Contains 154 type 4.2 pins and 63 mixed oxide blanket pins. Had a fission gas leak at beginning of life (possible end cap weld failure).
69C #6	16505 TDFA 16523 TDFA 8252 FEX 2144 FEX 2065 FEX	Pin container storage of miscellaneous pins	Fuel returned from INL, consists of: 39 pins of D9-2 26 pins of D9-4 11 pins of ACO-1 12 pins of ACO-3 - Added @ FFTF 15 pins of PO-1
69C #34	4123 TDFA 16527 DE-9 2143 FEX 2155 FEX 2065 FEX	Pin container storage of miscellaneous pins	Fuel returned from INL, consists of: 36 pins of RTCB-7 40 pins of DE-9 (one encapsulated DNM pin) 2 pins of FO-2 34 pins of MFF-1A 2 pins of PO-1 (2 encapsulated gas leaker pins)
69D #50	2144 FEX 2158 FEX 2170 FEX 2171 FEX 2172 FEX	Pin container storage of miscellaneous pins	9 pins of ACO-3 49 non metal pins of MFF-1 18 pins of MFA-1 12 pins of MFA-2 10 pins of MBA-1
69D #51	2158 FEX	Pin container storage of MFF-1 pins	114 non metal pins of MFF-1
69D #52	2069 FEX	Pin container storage of PO-4 pins	141 fuel pins from PO-4
69D #55	2069 FEX	Pin container storage of PO-4 pins	28 fuel pins from PO-4, including 6 suspect DNM leaker pins placed into failed fuel pins storage tubes.
69E #60	2075 FEX	Highly Enriched	SRF-3 Inner Bundle containing highly enriched uranium test pins.
69E #61	2076 FEX	Highly Enriched	SRF-4 Inner Bundle containing highly enriched uranium test pins.

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Revision 1

Attachment 1: Return of Fast Flux Test Facility Experiments and Repackaged Fuel

FFTF-17294

Revision 1

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Revision 1

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- References: (1) Internal Memo, R. L. Simons to L. R. Besel, "Radiation Source Distribution for the FFTF Off-Load Program," RLS-91-008, dated October 10, 1991.
- (2) Internal Memo, R. L. Simons to L. R. Besel, "Revised Radiation Source Distribution for the FFTF Fuel Off-Load Program," RLS-92-004, dated May 4, 1992.

Fast Flux Test Facility (FFTF) Safety evaluated the adequacy of the radiation shielding provided by the Bottom Loading Transfer Cask (BLTC), Solid Waste Cask (SWC), the Interim Storage Cask (ISC), and other fuel handling stations for handling and storing irradiated fuel from experimental assemblies ACO-3, ACO-1, FO-2, MFA-1, MFA-2, AAD-5, DE4-1, D9-1, DEA-2 and C-1. It is concluded that the radiation shielding provided by the BLTC, SWC, and the ISC is adequate for handling and storage of the proposed fuel. This conclusion is based on the fact that the radiation source associated with an Ident 69 with a maximum of 4 kg of the experimental fuel is less than the radiation source used as the basis for the original design of the BLTC, SWC, and the ISC radiation shielding (References 1 and 2).

The radioactive inventory of 4 kg of the experimental fuel would be less than the inventory of a typical worst case driver fuel assembly which contains about 36 kg of irradiated fuel and many kg of activated cladding and fuel duct material. The fact that some of the experimental fuel has a higher enrichment, a maximum of 31.2% versus the 29.3% for the design basis case, and has a higher burn up, a maximum of 231,500 MWd/MTHM versus 150,000 MWd/MTHM, could increase the radiation source term, on a per kg basis, by a factor of about 1.64. However, this would not make up for the 4 kg of experimental fuel versus the 36 kg in the design basis driver fuel assembly (DFA). In addition, the experimental fuel will be placed in unirradiated tubes and then into an unirradiated closed container and as such will contain no or very little irradiated stainless steel and the associated Co-60. For radiation shielding for the times periods of interest here, Co-60 is the dominant isotope. Spacers will be used to locate the closed spent fuel container at the same axial positions as a DFA inside an Ident 69.

kmc

FFTF-17294
Revision 1

**Attachment 2: Interim Storage Cask Fuel Pin Temperature Limit Assessment for
Specialized FFTF Test Assemblies**

FFTF-17294
Revision 1

CORRESPONDENCE DISTRIBUTION COVERSHEET

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Subject: INTERIM STORAGE CASK FUEL PIN TEMPERATURE LIMIT ASSESSMENT FOR
SPECIALIZED FFTF TEST ASSEMBLIES

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March 17, 1997

LMHC-9751842

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Post Office Box 1050 MSIN H0-35
Richland, Washington 99352

Dear Mr. Dobbin:

**INTERIM STORAGE CASK FUEL PIN TEMPERATURE LIMIT ASSESSMENT FOR SPECIALIZED
FFTF TEST ASSEMBLIES**

- References:
- 1) Letter, B. C. Gneiting to L. R. Besel, "Pin Temperature Limits to Restrict Stress Rupture Breaches During Long Term Storage of FFTF Fuel Assemblies," dated January, 27, 1992.
 - 2) *FFTF Final Safety Analysis Report*, WHC-TI-75001, Amendment 76, January 1996.
 - 3) Letter, B. C. Gneiting to S. Guttenberg, "Long Term Storage of FFTF Fuel Assemblies - Fuel Pin Temperature Limits," dated March 27, 1996.
 - 4) D. L. Nielsen, *Review of FFTF Fuel Experiments for Storage at ISA*, WHC-SD-FF-RPT-005, Rev. 1, October 18, 1996.
 - 5) Letter, A. E. Bridges (DESH) to B. C. Gneiting (LMHC), "Fast Flux Test Facility Status and Information," DESH-9751442, dated February 19, 1997.

An analysis of the stress rupture behavior of Fast Flux Test Facility (FFTF) irradiated fuel assemblies contained in Interim Storage Casks (ISC) for up to 50 years was previously performed in Reference 1. The analysis in Reference 1 was used in developing a maximum initial fuel pin temperature limit of 900°F for ISC storage (Reference 2). The analysis presented in Reference 1 covers FFTF driver fuel assemblies and similar experimental assemblies containing mixed-oxide fuel irradiated in FFTF (see Reference 3). The purpose of the analysis presented in this letter is to address other FFTF irradiated test assemblies identified in Table 3 of Reference 4 plus the SRF assemblies. These other test assemblies requiring further analysis are categorized by the following four fuel types: 1) Blanket Assemblies, 2) Carbide Fuel, 3) Sodium Bonded Metal Fuel, and 4) Highly Enriched Fuel Assemblies.

The same analysis method defined in Reference 1 was used in this analysis for the four fuel types given above. A summary of the key assumptions are listed below. Additional information associated with the method, assumptions, and limitations can be obtained from Reference 1. The four

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fuel types in this analysis contain the same three cladding types (316 Stainless Steel (SS), D9, and HT-9) used in the original analysis. Therefore, the stress rupture equations used in this analysis are the same as defined in Reference 1. A time-to-rupture value is calculated from the stress rupture equations for a time period of constant stress and temperature. For this analysis, the ISC design limit for the maximum initial fuel pin temperature (900°F) was used. The affect of seasonal temperature changes is covered in the temperature limit calculation. A temperature versus time table was created using the decay heat versus time curve and the decay heat versus temperature curve from the last two references in Reference 1. For a 900°F initial temperature, the following temperature versus time table was developed:

Time (yr)	Temp. (°F)
0.00	900
0.06	888
0.31	850
0.81	780
1.31	716
2.31	662
3.81	615
6.31	546
11.31	485
16.31	464
26.31	431
36.31	410
50.00	386

Cladding stresses were calculated using the standard thin wall formula for a tube with an internal gas pressure. The stress equation uses the tube midwall diameter and the thickness. To obtain a conservative stress value, the cladding thickness was reduced by 5 mils to account for some wastage. Fuel pin pressures were adjusted for the change in temperature using the ideal gas law.

The information contained in Reference 5 was collected to support analyses of the test assemblies categorized by the three fuel types in Reference 4. The data obtained from Reference 5 to perform the analyses are cladding dimensions and temperatures and plenum pressures for a given cladding type. To perform the analyses for the fourth fuel type, the same type of data was obtained from the independent safety evaluation analysis. Similar tests were grouped for each fuel type and a bounding case was selected using the end-of-life (EOL) design conditions. Values used for the key parameters were selected so that a conservative result would be produced. Some examples are: using EOL design conditions; using absolute rather than gauge pressure with no external pressure on the cladding; using midwall temperatures, when available, rather than inner surface temperatures so the pressure at 900°F will be higher.

Test assembly group definition, bounding values used, parameter adjustments, and maximum pressure and stress values are presented below for each of the four fuel types.

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Fuel Type 1: Blanket Assemblies Uranium Oxide/Thorium Oxide

The blanket tests were analyzed in four groups based on cladding type and size and EOL conditions. The four groups are: six ABA assemblies (1 - 6), two WBA assemblies (40 & 41), the UO-1 assembly, and the MBA-1 assembly. The EOL design conditions for the ABA-2 test assembly were used as the bounding conditions for the ABA series analysis. A plenum pressure was not available for the WBA series. The same plenum pressure and cladding temperature used for the ABA series was used for the WBA series since the plenum volume and the irradiation conditions were bounded by the ABA series. The UO-1 test was an enriched uranium oxide fuel experiment designed to test several fuel and cladding variables. It was also designed to be prototypic of an FFTF Series IV driver fuel assembly. This assembly should be bounded by the analyses in Reference 1, but is repeated here using the worst case irradiation conditions in a 316 SS tube. The EOL design conditions were used for the MBA-1 calculations. The AB-1 experiment had a very low burnup and is similar to a driver fuel assembly. Therefore, the AB-1 assembly is bounded by the analyses in Reference 1. Key parameters defined for the Type 1 test set are given below.

Test Set:	ABA	WBA	UO-1	MBA-1
Cladding Type:	HT-9	316SS	316SS	D9
Cladding Thickness, in:	0.020	0.015	0.015	0.20
Cladding OD, in:	0.390	0.506	0.230	0.457
Burnup, MWD/kgM:	58	58	70	25
Cladding Temperature, F:	1204	1204	1123	1258
Plenum Pressure, psia:	446	446	599	162
Adjusted Clad Thickness, in:	0.015	0.010	0.010	0.015
Adj. Clad Midwall Dia., in:	0.375	0.496	0.220	0.442
Plenum Press. @ 900°F, psia:	365	365	512	128
Clad Hoop Stress @ 900°F, psi:	4556	9075	5682	1889

Fuel Type 2: Carbide Fuel

The three carbide experiments (ACN-1, FC-1, and AC-3) are similar and were grouped together for this analysis. The bounding conditions used for this set were obtained from the ACN-1 EOL design conditions using the thinnest cladding and the maximum plenum pressure. Two cladding types were used in the assembly, 316 SS and D9. Stress rupture calculations are very similar for 316 SS and D9. The 316 SS cladding type was selected for use in this calculation. Key parameters for the Type 2 test set are given below.

Cladding Type:	316 SS
Cladding Thickness, in:	0.015
Cladding OD, in:	0.370
Burnup, at%:	8.6
Cladding Temperature, F:	1115
Plenum Pressure, psia:	470
Adjusted Clad Thickness, in:	0.010
Adj. Clad Midwall Dia., in:	0.360
Plenum Pressure @ 900°F, psia:	406
Clad Hoop Stress @ 900°F, psi:	7335

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Fuel Type 3: Sodium Bonded Metal Fuel

The sodium bonded metal fuel experiments were analyzed in two groups; seven MFF assemblies (1 ~ 6, & 8A) and the IFR-1 test. The bounding conditions used in the MFF series analysis were obtained from the EOL design conditions for the MFF-3 experiment. The EOL design conditions were used for the IFR-1 analyses. Key parameters for the Type 3 set are given below.

Test Set:	IFR-1	MFF
Cladding Type:	D9	HT-9
Cladding Thickness, in:	0.022	0.022
Cladding OD, in:	0.270	0.270
Burnup, MWD/kgM or at%:	10 at%	183
Cladding Temperature, F:	1033	1078
Plenum Pressure, psia:	1288	1491
Adjusted Clad Thickness, in:	0.017	0.017
Adj. Clad Midwall Dia., in:	0.253	0.253
Plenum Pressure @ 900°F, psia:	1173	1318
Clad Hoop Stress @ 900°F, psi:	8730	9811

Fuel Type 4: Highly Enriched Fuel Assemblies (HEFA)

The two HEFA experiments (SRF-3 & -4) are essentially the same and were grouped together for this analysis. The bounding conditions used for this set were obtained from the SRF-3 nominal steady-state performance calculations for EOL design conditions. The SRF fuel pins were the same as a standard driver, except for fuel enrichment and the use of blankets in place of the reflectors. These two tests would have been covered in the analysis in Reference 1 if the top blanket had not reduced the effective plenum length by 30%. The reduced plenum volume resulted in higher plenum pressures than used in the Reference 1 analyses. Key parameters for the Type 4 set are given below.

Test Set:	SRF-3
Cladding Type:	316 SS
Cladding Thickness, in:	0.015
Cladding OD, in:	0.230
Burnup, MWD/kgM:	63
Cladding Temperature, F:	1089
Plenum Pressure, psia:	720
Adjusted Clad Thickness, in:	0.010
Adj. Clad Midwall Dia., in:	0.220
Plenum Pressure @ 900°F, psia:	632
Clad Hoop Stress @ 900°F, psi:	6980

An estimate of the cladding damage was determined using a life fraction rule that accumulates fractional damage linearly. Damage fractions for each time period of constant stress and temperature were calculated and accumulated by summing the ratio of each time increment over the time-to-rupture. This summed damage fraction is called a Cumulative Damage Fraction (CDF) value. The CDF values calculated for each test set analyzed is given below:

Test Set:	ABA	WBA	UO-1	MBA-1	Carbide	IFR-1	MFF	HEFA
CDF Value:	4.2E-12	3.6E-10	9.9E-5	4.9E-6	2.0E-4	3.3E-4	1.2E-9	1.7E-4

(Note: E-4 represents 10 raised to the -4 power)

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The same CDF value ($1.7E-3$) used in the Reference 1 analysis as an upper limit is assumed to apply for this analysis. Use of this CDF limit and the conservatism used in the analysis should achieve a reasonably low probability of breach from stress rupture due to gas pressure loading. All the CDF values calculated in this analysis are below this limit.

In summary, the following test assemblies listed in Table 3 of Reference 4 as needing further evaluation and the HEFA experiments are covered by the ISC design fuel pin temperature limit of 900°F.

Fuel Type 1, Blanket Assemblies

ABA-1, ABA-2, ABA-3, ABA-4, ABA-5, ABA-6, WBA-40, WBA-41
MBA-1, UO-1, & AB-1

Fuel Type 2, Carbide Fuel

FC-1, AC-3, & ACN-1

Fuel Type 3, Sodium Bonded Metal Fuel

MFF-1, MFF-2, MFF-3, MFF-4, MFF-5, MFF-6, IFR-1, & MFF-8A

Fuel Type 4, Highly Enriched Fuel Assemblies

SRF-3 & SRF-4

The results indicate a low probability of breach from stress rupture.

Very truly yours,

B. C. Gneiting, Principal Engineer
Lockheed Martin Hanford Corporation

bab

Reviewed By:

S. L. Hecht
Duke Engineering & Services Hanford, Inc.

Date: 3-20-97

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Attachment 3: Placing Delayed Neutron Signal Breached Fuel Pins into Long Term Storage

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Fluor Hanford
Post Office Box 1000
Richland, Washington 99352

FLUOR

Memorandum

To: W. W. Bowen* N2-53 Date: May 13, 2003
S. A. Chastain* N2-02
S. W. Hiller* N2-02
D. L. Nielsen* N2-53
W. V. Witherspoon* N2-13

From: Original signed by T. M. Burke for Telephone 376-9779
W. F. Brehm, Lead ne:
Components & Surveillance
Engineering

cc: T. M. Burke* File No.: WWB-03-004
WWB File/LB
*electronic distribution

Subject: PLACING DELAYED NEUTRON SIGNAL BREACHED FUEL PINS INTO
LONG TERM STORAGE

- References:
- 1) WHC-SP-0980, Postirradiation Nondestructive Examination Data Report for FFTF High Power Assembly DE-9, dated 1993.
 - 2) WHC-SD-FF-DA-079, Rev. 0, Residual Gases in Interim Storage Cask/Core Component Container, dated 1995.

Situation

The question to be answered is "what is necessary to place delayed neutron (DN) signal breached fuel pins into long-term storage in an Interim Storage Cask (ISC) with confidence that there will be no release of radioactive material, particularly fuel particles, or buildup of flammable gases inside the ISC?"

Consider what happens to a fuel pin when it is breached and the reactor is operated until a DN signal is received. The reactor is then shut down so that the fuel pin is not emitting any more fission gas.

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The presence of a DN signal usually means that there has been fuel-to-sodium contact and that sodium may have entered the fuel pin through the breach. How much sodium has entered the pin is not always easy to quantify. Shutting down the reactor means that no more fission gas will be emitted from the fuel pin. The pressure of the flowing sodium past the breach site then is higher than the fission gas pressure inside the pin, and thus sodium could enter the inside of the pin through the breach. The sodium-fission product mixture present at the breach site (sodium-cesium and perhaps other fission products) is thought to be extremely corrosive to fuel cladding at reactor temperatures, and the breach can conceivably open further as a result of this corrosion.

Removing the fuel pin from the high-temperature environment will slow the sodium ingress and corrosion at the breach site. However, it is possible that the pin will be left with some sodium inside. However, it is not easy to quantify how much sodium is inside the pin without disassembling the fuel assembly and weighing the pin.

Fuel assemblies containing known breached pins are not put through the conventional wash cycle in the Sodium Removal System because of criticality safety concerns documented in the Fast Flux Test Facility (FFTF) Final Safety Analysis Report Section 20.4.2.D.1.4). Also, the water present in the rinse cycle could react with the sodium at or inside the breach, creating a very corrosive sodium-cesium hydroxide mixture. This mixture could corrode the stainless steel cladding and further open the breach. Present plans are to disassemble the DN signal assemblies, identify and individually wipe down each breached pin, encapsulate them in an outer protective stainless steel tube, and then store them in an "Ident-69" or similar container for placement into an ISC. The question remains as to proper preparation and storage conditions of these pins for extended periods, tens to hundreds of years, or even longer. It is noted that fewer than five fuel assemblies contain known or suspected DN leakers.

Photographs of a breached pin from the DE-9 test assembly, taken with a scale alongside the breach, showed the opening of the breach to be approximately 0.6 inch long by 0.05 inches at the widest spot. It is conceivable that several grams of sodium could enter through the breach location. However, weight change data on the breached pin was in good agreement with all other pins from the assembly (Reference 1), indicating that little, if any, sodium was left inside the pin.

Because of the limited amount of water from the damp rags used to wipe the pin down, or with the small amounts of moisture that are postulated to leak into the ISC/ Core component container (CCC), those circumstances will not create significant amounts of sodium-water reaction.

The amount of sodium that could conceivably be present inside a breach pin, if all the void space inside the pin was filled with sodium as a result of a breach, is estimated to be approximately 23 grams. This estimate is obtained by assuming that in an 8 foot long fuel pin, 4 feet of that interior space is occupied by fuel pellets, shield pellets, springs, etc, then calculating the volume of a 4 foot long x 0.200 inch cylinder (24.73 cc) and multiplying by the density of sodium. Based

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on experience with DE-9, gram quantities of sodium may not be present in breached pins. The breached pins can be identified and weighed to within one gram precision. Therefore, if a DN pin(s) is located with several grams of sodium, it can be evaluated on a case-by-case basis by Engineering prior to approval for storage in an ISC.

Discussion

Several choices exist for dealing with a fuel pin containing a breach that resulted in a DN signal, among them are the following:

1. Ignore the presence of the breach and wash the fuel assembly by the normal process. This is not acceptable for reasons discussed above and below.
2. Remove the affected pin from the fuel assembly and encapsulate it without removing any of the adherent sodium. This step is not compatible with the goal of minimizing or completely eliminating the presence of residual unreacted sodium inside the CCC.
3. Remove the affected pin from the assembly, carefully wipe it down with a damp cloth or similar device, then encapsulate. The encapsulation tube will have threaded, not welded, closures so that some slow ingress and egress of gas could still occur.

Process #3 is the present reference process. Performing that process will minimize the surface area of unreacted sodium inside the ISC.

There could be some slow oxidation of the sodium at the breach site by the residual oxygen and water vapor in the inert gas inside the CCC, but with the low level of impurities and the extremely small cross-section of the breach, there will not be any significant reaction. The CCC is expected to have initially, and maintain, very low levels of oxygen and water vapor (200 ppm or less oxygen and water vapor in the Interim Examination and Maintenance [IEM] Cell atmosphere).

Under these conditions, the reaction of any sodium at the breach site with water vapor would probably produce some small amount of anhydrous sodium hydroxide. Reference 2 assumed that there were 2 grams of unreacted sodium inside a CCC. The likely sequence over 50 years of storage is that the small amount of oxygen and water vapor inside the ISC, plus the small amount of air that leaks in, would form at first anhydrous sodium hydroxide with exposed sodium. This hydroxide would likely remain as a solid essentially anhydrous compound. Anhydrous sodium hydroxide melts at 605°F so that it would tend to remain in place rather than drain away. Further reaction with water vapor would be slowed or completely stopped by this anhydrous sodium hydroxide layer. If there is sufficient air inleakage, the water vapor and carbon dioxide in the air will convert the anhydrous sodium hydroxide to first an aqueous sodium hydroxide solution; then

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to sodium carbonate/bicarbonate. This carbonate/bicarbonate mixture forms a solid coating on the sodium surface, and again slows down or completely stops the reaction of the metallic sodium. (The water vapor/carbon dioxide reaction is the basis for the "passivation" process being proposed for stabilizing sodium residuals in FFTF piping and components.)

As a bounding case, consider the amount of sodium that would have to be present, and reacted with water, to produce a flammable mixture inside the ISC/CCC, assuming two extremely unlikely conditions, first that the inert atmosphere inside the ISC/CCC is replaced with air, and second, that sufficient water can leak in.

The volume of a CCC/ISC with seven fuel assemblies is 609,000 cubic centimeters (cc) (Reference 2). The lower flammable limit of hydrogen in air is 4% (4% of 609,000 is 24,360 cc). If one mole of gas is 22,400 cc, then the amount of hydrogen necessary to make a flammable mixture in an ISC/CCC filled with air is 1.09 moles.

From the chemical equation $\text{Na} + \text{H}_2\text{O} \Rightarrow \text{NaOH} + \frac{1}{2} \text{H}_2$, one mole of hydrogen is produced by reacting two moles of sodium with water. Therefore 1.09 moles of hydrogen will react with 2.18 moles of sodium.

The gram-molecular weight of sodium is 23, so $2.18 \text{ moles} \times 23 \text{ grams/mole} = 50.14 \text{ grams}$ of sodium would need to react with water to produce a flammable gas mixture in air.

Reference 2 discusses production of hydrogen by other means, such as radiolysis of the water vapor present in the gas when the ISC/CCC is filled, water vapor that could leak in, and radiolysis of the BUNA-N seals. That amount of hydrogen is between 100 and 200 cc, far less than the 24,360 cc of hydrogen discussed above. Of course, if one assumes that an air atmosphere is present, there will be far more than 200 ppm water vapor present and the radiolysis calculations must be repeated. Reference 2 indicated that up to 98.8 cc of hydrogen could be produced due to hydrolysis over the assumed 50 year storage life based on a water vapor fraction of 0.45% (or 99.5 cc of water leaking into the ISC/CCC). If an air atmosphere is assumed with 0.45% water vapor fraction, $609,000 \times 0.0045$ results in 2740 cc of water vapor initially present which results in 2720 cc of additional hydrogen present. This would still be far less than the amount of hydrogen discussed above. The ISC provides the capability for obtaining a gas sample prior to opening the cask at the end of the storage period which alleviates concerns regarding unexpected venting of flammable gases when the cask is opened. The estimated amount of flammable gas for this unlikely scenario would be well below the lower flammability limit.

If the ISC boundary is maintained intact through the storage life, then storing the breached pins with no additional confinement or containment, other than the already planned encapsulation, would not be required. If, on the other hand, sufficient water entered the ISC and the CCC to reach the breach location, a sodium-water reaction inside the pin could occur. The reaction of

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sodium, water, and some of the fission products such as cesium is thought to produce a very corrosive mixture, which would corrode the stainless steel at the breach location and enlarge the size of the breach and loss of fuel to the outside of the pin could not be ruled out. However, ingress of sufficient amounts of water to cause sodium-water reactions inside the pin is considered to be incredible. The ISC has stringent design criteria and is constructed to ensure sealed dry storage of the fuel is maintained. With that condition, and the likelihood that little sodium-water reaction would occur from residual moisture inside the ISC/CCC under expected storage conditions, it can be concluded that the potential for the significant sodium-water reaction producing damage to the fuel pin and release of fuel material is not credible.

The present plan to enclose the breached pin in another closed tube after wiping down adds further conservatism. It is noted that the outer tube will have threaded connections which are not to be considered leak tight. This situation will permit any hydrogen formed by sodium-moisture reactions to escape the confines of the tube. It should be verified that heat transfer from the tube will be sufficient to preclude excessive temperatures in the enclosed pin.

In summary, the low probability of any significant amount of sodium inside breached pins, ability to detect any such sodium if it were present, added to the planned encapsulation of breached pins, the inert atmosphere inside the ISC/CCC, and the capability to obtain samples of the cask atmosphere prior to opening provide sufficient conservatism and defense-in-depth to conclude that no additional measures are required to mitigate the potential for sodium reactions inside the ISC/CCC.

Acknowledgments

I would like to acknowledge Debbie Nielsen, Peggy Scott, JR Vincent, Ed McPherson, and Steve Chastain for their assistance preparing this data.

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Attachment 4: Fast Flux Test Facility Status and Information

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DE&S Hanford, Inc.

A Duke Engineering & Services Company

P.O. Box 350
Richland, WA 99352

February 19, 1997

DESH-9751442

B. C. Gneiting
Lockheed Martin Hanford Corporation
Post Office Box 1500
Richland, Washington 99352-1505

Dear Mr. Gneiting:

FAST FLUX TEST FACILITY STATUS AND INFORMATION

- References: (1) D. L. Nielsen, *Review of FFTF Fuel Experiments for Storage at ISA*, WHC-SD-FF-RPT-005, Rev. 1, October 18, 1996.
- (2) Letter; A. E. Waltar, WHC, to G. Muhling, EFR, "Action on Specific Areas of Consensus Items from May 1992 Exchange Meeting," 9255030, dated July 2, 1990.

Per your request this letter documents the additional information on Fast Flux Test Facility (FFTF) test assemblies, identified in Table 3 of Reference 1. These tests were noted as requiring further analysis prior to storage in Interim Storage Casks. Specifically, the three (3) fuel categories of Sodium Bonded Metal Fuel, Carbide Fuel, and Blanket Assemblies were identified as requiring analysis of their design fuel pin temperature limits. Per your request, end-of-life (EOL) plenum pressure, and cladding midwall temperature are provided for the tests in each of the above fuel categories. This activity was funded by the Task Package Charge Number (TPCN) B193D. It is assumed any required peer reviews will be coordinated by you. The primary source of these data were technical files and documents maintained by the Spent Nuclear Fuel Evaluations group.

Table 1 of Reference 2 was used as the starting point for summarizing the requested test assembly information. This table lists the design, operational, and status as of 1992 for FFTF irradiation tests. Two additional columns of information were added to provide beginning-of-life (BOL) and EOL design conditions for the test assemblies of interest. The test information is summarized in Attachments 1 through 5 and described in the following sections. This information was obtained primarily from the safety analysis report for each test. Pertinent pages from these documents are provided for your information in Attachment 6.

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Attachment 1, lists the desired design information for the Sodium-Bonded Metal Fuel tests: IFR-1, MFF-1, MFF-2, MFF-3, MFF-4, MFF-5, MFF-6, and MFF-8A. The maximum cladding temperature for the IFR-1 and MFF-1 tests are at the cladding inner surface; however, the maximum cladding temperature for the other tests is at the cladding midwall. Tests MFF-5 and MFF-6 are bounded by the MFF-4 test. Please note that the MFF-8A test was not irradiated in FFTF and should be treated as fresh fuel.

Attachment 2, lists the desired design information for the Carbide Fuel tests: ACN-1, FC-1, and AC-3. Safety analysis reports were available only for the ACN-1 and AC-3 tests. However, there is a statement in the AC-3 documentation that states that the FC-1 and AC-3 tests were designed to operate under similar conditions. If it is assumed that the FC-1 and AC-3 cladding temperatures at EOL are the same, then the AC-3 analysis should bound the FC-1 test. If it is necessary to obtain more information on the FC-1 test, the safety analysis report should be ordered or the Irradiations Test Management (ITMA) file for FC-1 should be retrieved from the Records Center. The maximum cladding temperature for these tests is at the cladding midwall.

Attachment 3, lists the desired design information for the Blanket tests: WBA-40, WBA-41, ABA-1, ABA-2, ABA-3, ABA-4, ABA-5, ABA-6, and MBA-1. The only references readily obtainable for the WBA-40 and WBA-41 blanket tests were the Independent Safety Evaluation documents. Only limited information on each of the tests was available. The maximum cladding temperatures for the WBA-40 and WBA-41 tests are at the cladding outer surface; however, the maximum cladding temperatures for the other tests are at the cladding inner surface.

Attachment 4, lists the desired design information for the Uranium Oxide Fuel test: UO-1. This test consisted of several fuel types. Information for each fuel type is listed on Attachment 4. The maximum cladding temperature is at the cladding midwall.

Attachment 5, lists the desired design information for the Axial Blanket Fuel test: AB-1. No safety analysis report for this test was readily obtainable. Again, if more detailed information is required, this document could be ordered or the ITMA file could be retrieved from the Records Center. However, since this test's purposes was to test different axial blankets and its irradiation time was very short (34.1 EFPD), it may be possible to bound this test with an analysis of the FFTF Row 1 driver fuel pin.

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If you have any questions regarding this subject, please contact me at 376-1337.

Very truly yours,

Alvin E. Bridges
A. E. Bridges
Spent Nuclear Fuel Evaluations

Jmn

Attachments 6

FDNW - K. D. Dobbin*

*Attachments 1 through 5

CONCURRENCE:

R. B. Baker
R. B. Baker
Duke Engineering & Services Hanford, Inc.

Date: 2/19/97

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ATTACHMENT 1

Table 1

**Consisting of 4 pages,
including cover page**

Table 1. Status of T41T Irradiation tests.

Program: Metal Fueled Tests

Test Designation	Assembly Type	Test Purpose and Variables	Conditioning	Feed or Blend	Element or Pb	Mixed Oxidizing Conditions	Assembled Exposure Through Non 1995	Beam Area 1995	MOI Dose/Condition	MOI Dose/Condition
• Test Name • JTNA • M2-M6	• Random • Dose Material • No. of Test		• Type and Condition • O.D., mm • Thickness, mm	• Type • Composition • Film Density, STD • Diameter, mm • Oxygen/Metal Ratio	• Element Oxy, mm • Element Feed Density, STD • Pb 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100, 101, 102, 103, 104, 105, 106, 107, 108, 109, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132, 133, 134, 135, 136, 137, 138, 139, 140, 141, 142, 143, 144, 145, 146, 147, 148, 149, 150, 151, 152, 153, 154, 155, 156, 157, 158, 159, 160, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170, 171, 172, 173, 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774, 775, 776, 777, 778, 779, 780, 781, 782, 783, 784, 785, 786, 787, 788, 789, 790, 791, 792, 793, 794, 795, 796, 797, 798, 799, 800, 801, 802, 803, 804, 805, 806, 807, 808, 809, 810, 811, 812, 813, 814, 815, 816, 817, 818, 819, 820, 821, 822, 823, 824, 825, 826, 827, 828, 829, 830, 831, 832, 833, 834, 835, 836, 837, 838, 839, 840, 841, 842, 843, 844, 845, 846, 847, 848, 849, 850, 851, 852, 853, 854, 855, 856, 857, 858, 859, 860, 861, 862, 863, 864, 865, 866, 867, 868, 869, 870, 871, 872, 873, 874, 875, 876, 877, 878, 879, 880, 881, 882, 883, 884, 885, 886, 887, 888, 889, 890, 891, 892, 893, 894, 895, 896, 897, 898, 899, 900, 901, 902, 903, 904, 905, 906, 907, 908, 909, 910, 911, 912, 913, 914, 915, 916, 917, 918, 919, 920, 921, 922, 923, 924, 925, 926, 927, 928, 929, 930, 931, 932, 933, 934, 935, 936, 937, 938, 939, 940, 941, 942, 943, 944, 945, 946, 947, 948, 949, 950, 951, 952, 953, 954, 955, 956, 957, 958, 959, 960, 961, 962, 963, 964, 965, 966, 967, 968, 969, 970, 971, 972, 973, 974, 975, 976, 977, 978, 979, 980, 981, 982, 983, 984, 985, 986, 987, 988, 989, 990, 991, 992, 993, 994, 995, 996, 997, 998, 999, 1000	• 100 KPMQ/M • 1200 100 90 • 17.3, 17.3, 17.3 • 664.9 • Sound to D20	• 100 KPMQ/M • 1200 100 90 • 17.3, 17.3, 17.3 • 664.9 • Sound to D20	• 100 KPMQ/M • 1200 100 90 • 17.3, 17.3, 17.3 • 664.9 • Sound to D20		
• DPA-1 • AK181 • GAN, T-60 • M3 16492	• Random • DPA-1 • 160 Pm	• Test of Mixed Feed • Random Feed (DPA-1) and • Feed density using DPA-1 • and from feed type • There are 125 plus with • U-22, 15 plus • U-199-22, and 19 plus • with U-199-22.	• DPA-1, 160 Pm CV • 160 Pm •							

Table 1. Status of FFTF Irradiation Tests.

Program: Metal Fueled Tests

Test Designation	Assembly Type	Test Program and Variables	Charging	Fuel or Number	Element or Pin	Maximal Operating Conditions	Accumulated Exposure Through Test 1979	Status	RTS Design Conditions	RTS Design Conditions
• RTMA • HS-Ha	• Medium • Dist. Mashed • Dist. of Pin		• Type and Condition • G.D., mm • Thickness, mm	• Type • Compositions • Form • Pin Diameter, STD • Pin Length, mm • Other	• Elemental Comp. and • Isotopic Fuel Density, STD • Pin Spacing, P, mm • W, D, Pitch, mm • Pin Length, mm • Other	• Medium Powering • Gas Pressure 110 • Medium Chilling • Temp., °C • Medium Lower • Power, Watts	• Current Pumping • RTMA/AN • Power, E > 0.1 MW, 10 ⁴ atm • EPFD	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts
• RTF-3 • RTF-4 • HS 3168	• Medium • Dist. Mashed • Dist. of Pin	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• Medium Powering • Gas Pressure 110 • Medium Chilling • Temp., °C • Medium Lower • Power, Watts	• Current Pumping • RTMA/AN • Power, E > 0.1 MW, 10 ⁴ atm • EPFD	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts
• RTF-3 • RTF-4 • HS 3168	• Medium • Dist. Mashed • Dist. of Pin	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• Medium Powering • Gas Pressure 110 • Medium Chilling • Temp., °C • Medium Lower • Power, Watts	• Current Pumping • RTMA/AN • Power, E > 0.1 MW, 10 ⁴ atm • EPFD	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts
• RTF-3 • RTF-4 • HS 3168	• Medium • Dist. Mashed • Dist. of Pin	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• RTF-3 • RTF-4 • HS 3168	• Medium Powering • Gas Pressure 110 • Medium Chilling • Temp., °C • Medium Lower • Power, Watts	• Current Pumping • RTMA/AN • Power, E > 0.1 MW, 10 ⁴ atm • EPFD	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts	• Medium Chilling • Temp., °C (°F) • Medium Lower • Power, Watts

TABLE 1. STATUS OF FFPF IRRADIATION TESTS.

Program: Metal Fueled Test

Test Description	Assembly Type	Test Purpose and Variables	Chilling	Fuel or Reactor	Element or Pin	Measured Operating Conditions	Accumulated Exposure Through June 1972	Pin	FFTF Design Conditions
• Test Name • FFTF • M2-M6	• Reactor • FFTF • M2-M6	• Qualification Tests, ALJ • FFTF • FFTF	• Type and Condition • FFTF • FFTF • FFTF	• Type • FFTF • FFTF • FFTF • FFTF • FFTF • FFTF	• Discontinued Gas, see • Standard Fuel Density, STD • FFTF • FFTF • FFTF • FFTF • FFTF	• Reactor Position • Gas Phase ID • Maximum Chilling Temp. °C • Maximum Linear Power, W/cm	• Gas Exposure • Core Exposure • FFTF • FFTF • FFTF	• Pin • FFTF • FFTF	• Maximum Chilling Temp. °C • Maximum Linear Power, W/cm
• M2-M6 • M2-M6 • M2-M6	• Reactor • FFTF • M2-M6	• Qualification Tests, ALJ • FFTF • FFTF	• Type and Condition • FFTF • FFTF • FFTF	• Type • FFTF • FFTF • FFTF • FFTF • FFTF • FFTF	• Discontinued Gas, see • Standard Fuel Density, STD • FFTF • FFTF • FFTF • FFTF • FFTF	• Reactor Position • Gas Phase ID • Maximum Chilling Temp. °C • Maximum Linear Power, W/cm	• Gas Exposure • Core Exposure • FFTF • FFTF • FFTF	• Pin • FFTF • FFTF	• Maximum Chilling Temp. °C • Maximum Linear Power, W/cm
• M2-M6 • M2-M6 • M2-M6	• Reactor • FFTF • M2-M6	• Qualification Tests, ALJ • FFTF • FFTF	• Type and Condition • FFTF • FFTF • FFTF	• Type • FFTF • FFTF • FFTF • FFTF • FFTF • FFTF	• Discontinued Gas, see • Standard Fuel Density, STD • FFTF • FFTF • FFTF • FFTF • FFTF	• Reactor Position • Gas Phase ID • Maximum Chilling Temp. °C • Maximum Linear Power, W/cm	• Gas Exposure • Core Exposure • FFTF • FFTF • FFTF	• Pin • FFTF • FFTF	• Maximum Chilling Temp. °C • Maximum Linear Power, W/cm

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ATTACHMENT 2

Table 2

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Table 2. Status of FFTF Irradiation Tests.

Program: Other Fuel Tests - Carbide

Test Designation	Assembly Type	Test Purpose and Variables	Charging	Fuel or Blanket	Element or Pin	Postulated Operating Conditions	Accumulated Exposure Through June 1991	Status June 1992	BSL Disrupt Conditions	BSL Disrupt Conditions
• Test Name • IT24A • HS-46	• Rodtype • Dual Material • No. of Pins		• Type and Conditions • O.D., mm • Thickness, mm	• Type • Composition • Pins • Pin Diameter, STD • Diameter, mm • Oxygen/Metal Ratio	• Elementary Oxy. test • Standard Fuel Density, STD • Pin Spacing, P, mm • Wire, D, Pinch, mm • Fuel Length, mm • Other	• Reactor Position • Gas Phase 110 • Minimum Chilling Temp, °C • Minimum Linear Power, Watts	• Gas Exposure • Current Density, MA/AgM • Phase, S > 0.1 M/V, 10% above • E2770		• Minimum Chilling Temp, °C • Minimum Linear Power, Watts	• Minimum Chilling Temp, °C • Minimum Linear Power, Watts
• AC2-1 • LC134 • HS 3111	• Multi-Duct, Grid Spaced • 214 SS • 27 Carbide Pins, 90 MOE • Reactor Pins	• Diagnostic performance of carbide fuel and provides data supporting 3-year carbide fuel core synthesis.	• 214 SS, 2015 CW • 214 SS, 2015 CW • 214 SS • 6.300 No. Bunched • 6.300 No. Bunched	• Metal Carbide • Variable • Solid Pellets • 0.1 - 0.6 • 0.1275 • N/A	• 0.15 • 70 • Grid Spaced • N/A • 0.14 • 10 No. Bunched and 10 No. Bunched Pins	• 2144 • 407 • 0.09.5	• 0.00000424 • 70 • 12.6 • 416.6	• Irradiation Complete • Standard Pin at BSL • No PTE or Report • Standard to 7127	• 602 (1113) • 943 • 602 (1119) • 943	• 602 (1113) • 943 • 602 (1119) • 943 • 1.02 (223) Surface Bunched • 2.14 (413) Surface Bunched • 8.34 (413) Surface Bunched • 1.02 (223) Surface Bunched • 1.02 (223) Surface Bunched
• PC-1 • LC104 • HS 3068	• Rod, Rodtype • 214 SS • 91 Pins	• Diagnostic performance of carbide fuel and provides data supporting 3-year carbide core synthesis.	• 214 SS, 2015 CW • 214 SS • 9.396 • 6.300	• Metal Carbide • Variable • Solid Pellets • 0.1 - 0.6 • 0.1275 • N/A	• 0.15 • 70 • Grid Spaced • N/A • 0.14 • 10 No. Bunched Pins	• 2144 • 407 • 0.09.5	• 0.00000424 • 70 • 12.6 • 416.6	• Irradiation Complete • No PTE or Report • Standard to 7127	• 602 (1113) • 943 • 602 (1119) • 943	• 602 (1113) • 943 • 602 (1119) • 943 • 1.02 (223) Surface Bunched • 2.14 (413) Surface Bunched • 8.34 (413) Surface Bunched • 1.02 (223) Surface Bunched • 1.02 (223) Surface Bunched
• AC-3 • LC107 • HS 3071	• Rod, Rodtype • 214 SS • 91 Pins	• Diagnostic performance of carbide fuel and provides data supporting 3-year carbide core synthesis.	• 214 SS, 2015 CW • 214 SS • 9.396 • 6.300	• Metal Carbide • Variable • Solid Pellets • 0.1 - 0.6 • 0.1275 • N/A	• 0.15 • 70 • Grid Spaced • N/A • 0.14 • 10 No. Bunched Pins	• 2144 • 407 • 0.09.5	• 0.00000424 • 70 • 12.6 • 416.6	• Irradiation Complete • No PTE or Report • Standard to 7127	• 602 (1113) • 943 • 602 (1119) • 943	• 602 (1113) • 943 • 602 (1119) • 943 • 1.02 (223) Surface Bunched • 2.14 (413) Surface Bunched • 8.34 (413) Surface Bunched • 1.02 (223) Surface Bunched • 1.02 (223) Surface Bunched

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ATTACHMENT 3

Table 3

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Table 3. Status of FFTF Irradiation Tests.

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Table 3. Status of FFTF Irradiation Tests.

Program: Blanket Assembly Tests

Test Designation	Assembly Type	Test Program and Variables	Chaffing	Fuel or Blanket	Diameter or Pk	Mechanical Operating Conditions	Accumulated Exposure Through June 1992	Status	DOE Design Conditions	DOE Design Conditions
• Test Name • ITMA • HS-Ma	• Blanket • Dual Material • No. of Pins		• Type and Condition • G.D., mm • Thickness, mm	• Type • Composition • Fuel Density, g/cc • Diameter, mm • Oxygen/Water Ratio	• Diameter, mm • Round Fuel Density, g/cc • Fuel Density, g/cc • W.D., mm • Fuel Length, mm • Other	• Reactor Position • Gas Flow, l/min • Minimum Chaffing Temp, °C • Minimum Linear Power, W/cm • Pressure, MPa	• Cool Exposure • Power, MW • Pressure, MPa • EFTD		• Medium Chaffing • Temp, °C (°F) • Minimum Linear Power, W/cm • Pressure, MPa • Cool Exposure, MPa • Pressure, MPa • EFTD	• Medium Chaffing • Temp, °C (°F) • Minimum Linear Power, W/cm • Pressure, MPa • Cool Exposure, MPa • Pressure, MPa • EFTD
• AAA-3 • HS175 • HS 3142	• Dual Material • Amusement • HTF • 91 Pins	• Core demonstration • important (COB) loop • within lower blanket • used to simulate • HTF and optimized • design. See ACO fuel • test.	• HTF • 9.04 • 0.308	• UO ₂ • Depleted U • Fuel End Solid Pellets • 1.07 • 1.07 • 1.07	• 6.305 • 89 • 11.174 • 1.27, 132.6 • 134.3	• 3184 • — • 222.6	• 908 EFTD • 41 • 21.3 • 771.1	• Irradiation Complete • Cool Cycle MA • End Cycle 118.2 • No PE or Report • Status is P17	• 477 (910°) • 175 • Chaffing lower Surface	• 548 (1025°) • 374 • 1.42 (258) • 47 • 21.6 • Chaffing lower Surface
• AAA-4 • HS175 • HS 3142	• Dual Material • Amusement • HTF • 91 Pins	• Core demonstration • important (COB) loop • within lower blanket • used to simulate • HTF and optimized • design. See ACO fuel • test.	• HTF • 9.04 • 0.308	• UO ₂ • Depleted U • Fuel End Solid Pellets • 1.07 • 1.07 • 1.07	• 6.305 • 89 • 11.174 • 1.27, 132.6 • 134.3	• 3184 • — • 222.6	• 908 EFTD • 41 • 21.3 • 771.1	• Irradiation Complete • Cool Cycle MA • End Cycle 118.2 • No PE or Report • Status is P17	• 477 (910°) • 175 • Chaffing lower Surface	• 548 (1025°) • 374 • 1.42 (258) • 47 • 21.6 • Chaffing lower Surface
• AAA-5 • HS175 • HS 3144	• Dual Material • Amusement • HTF • 91 Pins	• Core demonstration • important (COB) loop • within lower blanket • used to simulate • HTF and optimized • design. See ACO fuel • test.	• HTF • 9.04 • 0.308	• UO ₂ • Depleted U • Fuel End Solid Pellets • 1.07 • 1.07 • 1.07	• 6.305 • 89 • 11.174 • 1.27, 132.6 • 134.3	• 3184 • — • 222.6	• 908 EFTD • 41 • 21.3 • 771.1	• Irradiation Complete • Cool Cycle MA • End Cycle 118.2 • No PE or Report • Status is P17	• 477 (910°) • 175 • Chaffing lower Surface	• 548 (1025°) • 374 • 1.42 (258) • 47 • 21.6 • Chaffing lower Surface
• AAA-6 • HS175 • HS 3145	• Dual Material • Amusement • HTF • 91 Pins	• Core demonstration • important (COB) loop • within lower blanket • used to simulate • HTF and optimized • design. See ACO fuel • test.	• HTF • 9.04 • 0.308	• UO ₂ • Depleted U • Fuel End Solid Pellets • 1.07 • 1.07 • 1.07	• 6.305 • 89 • 11.174 • 1.27, 132.6 • 134.3	• 3184 • — • 222.6	• 908 EFTD • 41 • 21.3 • 771.1	• Irradiation Complete • Cool Cycle MA • End Cycle 118.2 • No PE or Report • Status is P17	• 477 (910°) • 175 • Chaffing lower Surface	• 548 (1025°) • 374 • 1.42 (258) • 47 • 21.6 • Chaffing lower Surface

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Table 3. Status of FFTF Irradiation Tests.

Program: Blanket Assembly Tests

Test Designation	Assembly Type	Test Program and Variables	Chilling	Fuel or Blanket	Element or Pin	Blanket Operating Conditions	Assessment Exposure Through June 1992	Status June 1992	SOI Design Conditions	SOI Design Conditions
<ul style="list-style-type: none"> • Test Name • ITMA • HX-Rn 	<ul style="list-style-type: none"> • Workpiece • Blanket Model • HX-Rn 		<ul style="list-style-type: none"> • Type and Conditions • G.D., mm • Thickness, mm 	<ul style="list-style-type: none"> • Type • Composition • Pin • Pin Diameter, STD • Diameter, mm • Oxygen/Steel Ratio 	<ul style="list-style-type: none"> • Diameter Gap, mm • Standard Pin Density, STD • Pin Spacing, P, mm • Wire, B, Pinch, mm • Pin Length, mm • Other 	<ul style="list-style-type: none"> • Reactor Position • Gas Flow 110 • Maximum Chilling Temp, °C • Maximum Linear Power, W/cm 	<ul style="list-style-type: none"> • Gas Exposure • Current Density, $10^{16}/\text{cm}^2/\text{s}$ • Pressure, E > 0.1 MPa, 10% above • EPTD 		<ul style="list-style-type: none"> • Maximum Chilling Temp, °C (°F) • Maximum Linear Power, W/cm • Pressure, MPa (psi) • Gas Flow, $10^{16}/\text{cm}^2/\text{s}$ • Pressure, E > 0.1 MPa, 10% above 	<ul style="list-style-type: none"> • Maximum Chilling Temp, °C (°F) • Maximum Linear Power, W/cm
<ul style="list-style-type: none"> • M44-1 • HX18 • HX 2172 	<ul style="list-style-type: none"> • Blanket Model • Blanket Model • HX-Rn 	<ul style="list-style-type: none"> • Dimensions M44-1 • Blanket assembly • Irradiation performance and lifetime capability under steady state conditions relative to those expected in the proposed reactor. 	<ul style="list-style-type: none"> • Modified 316 • 11.000 • 0.300 	<ul style="list-style-type: none"> • UO₂ • Enriched U (3 wt %) • Pin End Solid Pellet • 97 • 10.340 • 2.00 - 2.02 	<ul style="list-style-type: none"> • 0.300 • 11.000 (Outer Pin) • 11.270 (Inner Pin) • 1.30, 2.1, (Outer Pin) • 0.30, 2.1, (Inner Pin) • 11.25 • - 	<ul style="list-style-type: none"> • 100 • 475 • 337.9 	<ul style="list-style-type: none"> • 740 EPTD for M44-1 • 31.8 • 11.02 • 0.61.5 	<ul style="list-style-type: none"> • Irradiation Conditions • Blanket Model M44-1 • Blanket Model M44-1 • No PTF for Support • Pin in PTF 	<ul style="list-style-type: none"> • 497 EPTD • 107.3 • 11.02 • Chilling Inner Surface 	<ul style="list-style-type: none"> • 481 (733°) at 100 EPTD • 301 • 11.02 • 0.61.5 • - • Chilling Inner Surface

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ATTACHMENT 5

Table 5

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Table 5. Status of FFTF Irradiation Test AB-1.

Test Designation	Assembly Type	Test Purpose and Variables	Chilling	Fuel or Moderator	Element or Pin	Heated Operating Conditions	Accumulated Exposure Through Test 1992	Time	BOC Design Conditions	EOC Design Conditions
<ul style="list-style-type: none"> Test Name FFTA HS-96 	<ul style="list-style-type: none"> Hardware Direct Shielded No. of Pins 		<ul style="list-style-type: none"> Type and Conditions O.D., mm Thickness, mm 	<ul style="list-style-type: none"> Type Composition Pins Pin Diameter, STD Diameter, mm Oxygen/Metal Ratio 	<ul style="list-style-type: none"> Elementary Gas, mm Elementary Fuel Density, STD Pin Length, P, mm Pin Length, mm Other 	<ul style="list-style-type: none"> Reaction Problem Gas Piping 119 Maximum Chilling Temp, °C Maximum Linear Power, Watts 	<ul style="list-style-type: none"> Good Exposure Current Exposure, MWd/kgM Pin, 8 > 8.1 MWd 10% (100%) EPD 	June 1992	<ul style="list-style-type: none"> Maximum Chilling Temp, °C (°F) Maximum Linear Power, Watts 	<ul style="list-style-type: none"> Maximum Chilling Temp, °C (°F) Maximum Linear Power, Watts Pin, 8 > 8.1 MWd, 10% (100%)
<ul style="list-style-type: none"> AB-1 GP044 HS 3114 	<ul style="list-style-type: none"> Std. Hardware with Removable Nozzle 316 SS 14 Removable 200 Pinout Pin 	<ul style="list-style-type: none"> Provide data on UO₂ and ThO₂ solid solution performance including OXN ratio and pin variations, Co-Pinout fuel 	<ul style="list-style-type: none"> 314 SS 30 S CV 3.94 8.25 	<ul style="list-style-type: none"> Metal Oxide Variable Enriched Pellets 97% M 4.35-4.99 3.96 	<ul style="list-style-type: none"> 8.14 84 7.28 1.422, 304.8 914.5 Add Reference 	<ul style="list-style-type: none"> 3300 412 472 	<ul style="list-style-type: none"> 95 MWd/kgM 8 1.8 34.1 	<ul style="list-style-type: none"> Pin Pinout Gas Leakage at BOC Leakage with Leak Removal from Core (Other ATFA) End Cycle 119 No PIS or Request Send to PTF 		

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**Attachment 5: Effect on Dose Consequences of Storing Metal Fuel Assemblies in the
Interim Storage Casks**

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FDNW-97-JCVK-001

January 22, 1997

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EFFECT ON DOSE CONSEQUENCES OF STORING METAL FUEL ASSEMBLIES IN THE INTERIM STORAGE CASKS

The purpose of this memo is to document the evaluation of the impact on the ISC dose consequences safety analysis of placing metal fuel assemblies in the ISC. Specific metal fuel assemblies to be stored in the Interim Storage Casks (ISC) are MFF-1 through MFF-6, IFR-1 and MFF-8A. The 18 sodium bonded pins for the carbide assembly ACN-1 are also considered. The conclusion of this evaluation is that the storage of the metal fuel assemblies or the 18 sodium bonded carbide pins in the ISC does not result in the dose calculations exceeding existing ISC radiological analyses, even for beyond design basis accidents. The bases for that conclusion are given in the attachment.

Sincerely,

John C. Van Keuren
Engineer
Safety Analysis & Risk Assessment

JCVK:gjr

Attachments (3)

Concurrence:

Brit E. Hey
Manager
Safety Analysis & Risk Assessment

Date 1/23/97

c: With Attachments

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to

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Standard driver fuel assemblies (DFAs) and Ident 69 pin containers are currently being off-loaded from the Fast Flux Test Facility (FFTF) and stored in Interim Storage Casks (ISC) at the Interim Storage Area (ISA) as authorized by the FFTF Final Safety Analysis Report (FSAR). For all design basis accident conditions evaluated, the storage system analyses demonstrate there will be no radiological releases. A beyond design basis and a hypothetical radiological release accident (WHC, 1995a and WHC, 1995b) were also analyzed for the DFAs to demonstrate the margin of safety associated with dry storage. These analyses concluded that the resulting dose consequences were within acceptable limits and that the FFTF fuel storage is not an unacceptable public health and safety hazard, even under the most extremely unlikely hypothetical event.

In addition to standard DFAs and pin containers, there are a number of different types of fuel and blanket experiments that have been irradiated in FFTF. A review of advanced oxide, helium bonded carbide, and nitride fuel experiments was conducted to determine if storage of these irradiated experimental assemblies in the ISC would require any special radiological accident release considerations (Van Keuren 1996). The conclusion was that the existing radiological accident analyses are bounding for the experimental assemblies. Metal fuel assemblies were not addressed in that review.

The purpose of this memo is to document the evaluation of the impact on the ISC safety analysis of placing metal fuel assemblies and sodium bonded carbide pins in the ISCs. Specific metal fuel assemblies to be stored in the ISCs are MFF-1 through MFF-6, IFR-1 and MFF-8A. The 18 sodium bonded pins for the carbide assembly ACN-1 are also considered. The conclusion of this evaluation is that the storage of metal fuel assemblies or the sodium bonded carbide fuel in the ISC will not increase the doses from any accident such that existing radiological analyses for the ISC are exceeded. The following are the bases for that conclusion:

1. The independent safety evaluations (ISEs) for typical metal fuel assemblies have been reviewed (WHC-1986 and WHC-1988). Both evaluations indicate that results of a cladding breach for these types of experiments have the same benign consequences as a breach in the oxide driver fuel. This cladding breach evaluation was done assuming the fuel pins were operating at full power with relative high fuel temperatures. Release of fuel material or fission products during the dry storage condition, without the thermal energy to drive diffusion of material, would be even less.
2. The metal fuel consists of a uranium-zirconium alloy or a plutonium-uranium-zirconium alloy clad with stainless steel, D-9, or HT-9. The fuel is a solid metal rod. The structural properties of the metal fuel in terms of resistance to shock is better than that of the ceramic fuel used for the FFTF driver fuel assemblies and other experiments.

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The analysis in WHC 1995a and 1995b considered the doses from hypothetical events that damage the fuel. The events are not expected to occur since they are beyond design basis events, but the events bound any potential releases. An increase in the release of volatile fission products from metal fuel could occur from these hypothetical events since the release fraction for volatile solids may be higher for metal fuel than oxide. The analysis in WHC 1995a and 1995b assumed a 5% release of volatile solids (e.g. ^{137}Cs) from the oxide fuel while the accidents for metal fuel in Himes 1989 assumed a 17% release of volatile solids, based on a release fraction for metal fuel given in the PRISM reactor PSAR. However the analyses in WHC 1995a and 1995b indicated that the dose from the transuranics contributed 90% of the dose. The transuranic release from metal fuel will be negligible since the ductile metal is much less likely to release fuel material than the more brittle (crushable) ceramic oxide fuel.

The total dose from the metal fuel will therefore be less than the oxide. The volatile solids only contribute about 6 to 8% of the dose reported in WHC 1995 and WHC 1995b. A factor of 17/5 or 3.4 increase in volatile solids release would result in a dose that is only 20% to 30% of the oxide dose. Since the dose from the transuranics from the metal fuel would be negligible, the total dose from a metal fuel assembly would be about 30% of that of the oxide. The dose from an assembly containing a mix of metal and oxide pins would be less than an assembly with only oxide pins. The doses reported in WHC 1995a and 1995b are more than an order of magnitude below the risk guidelines with oxide fuel and would be even lower with metal fuel. The dose consequences are therefore not increased by placing metal fuel in the ISC.

3. The metal fuel forms a low melting eutectic with the cladding at fuel cladding temperatures above approximately 700 °C (1300 °F). There is no mechanism for the fuel to reach this temperature during storage in the ISC since the decay heat of the assemblies is very low (<250 W). The thermal evaluation given in the Appendix H of the FFTF FSAR indicates that the maximum cladding temperature reached in any credible accident in the ISC is less than 480 °C (900 °F). This requirement applies to the metal fuel as well as other types of fuel. The dose consequences of thermal accidents will not be increased by placing metal fuel in the ISC.
4. The metal fuel has a sodium bond between the cladding and the fuel. The sodium is a potential combustion source, although the amount of sodium is small (7 to 8 g per pin). The beyond design basis event is based on a nonmechanistic crushing/shearing of fuel. The metal fuel swells during irradiation, pushing the sodium up into the plenum. A crushing in the fuel region would therefore not release significant amounts of sodium. A shearing in the plenum region could result in a release of sodium. If the sodium ignited, heat would be generated. However, even if all the sodium in the pins was released, all the sodium burned, and the heat contained adiabatically in the fuel assembly, the average

assembly temperature increase would be about 145 °C (See Attachment 2). The maximum fuel initial temperature is less than 427 °C. The temperature after the sodium burn would therefore be about 570 °C, which is 130 °C below the eutectic temperature. In fact the peak assembly temperature would be much lower since (1) a large fraction of the heat would be conducted to the container and cask, and (2) even if the sodium is assumed to ignite, only a fraction of the sodium would be exposed to air and burn.

It also should be noted that the eutectic penetration rate is very slow at or slightly above a temperature of 700 °C. For example, eutectic formation was noted in an experiment with U-19Pu-10Zr and D9 cladding only after the experiment was held at 725 °C for over 300 hours (Weber 1988). Thus, even if the fuel-cladding temperature exceeded the eutectic temperature for a short period of time, there would be no safety impact.

5. Most of the metal fuel is enriched uranium rather than a plutonium uranium mix. The plutonium is a significant contributor to the dose given in WHC 1995a and WHC 1995b. The dose from releases from the enriched uranium fuel material will be less than the mixed oxide fuel, even assuming the same amount of material is released.
6. There has been considerable handling and irradiation experience with metal fuel at the EBR-II test reactor in Idaho, which uses metal fuel as their driver assemblies. The metal fuel failure behavior has been benign. The fuel tends to release less gaseous fission products than oxide fuel during an in-reactor breach. (Tsai, 1986). The metal fuel does not react with the sodium coolant as does the oxide. There is no indication based on experience with metal fuel that the dose consequences from any credible accidents would be worse with metal fuel than with the oxide in storage.
7. The previous evaluation of storage of experiments in the ISC (Van Keuren, 1996) addressed the helium bonded carbide fuel, but did not address the sodium bonded pins. The ISE for the ACN-1 assembly has been reviewed (WHC, 1980). The sodium bond is a potential combustion source, but as discussed in item 4 above, the sodium quantities are low, and the sodium does not constitute a significant hazard. Carbide fuel does not form a low melting eutectic with the cladding, and the melting temperature for carbide fuel is over 2500 °C. Other considerations for carbide fuel are bounded by the oxide fuel analyses.

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Calculation of Temperature in a Metal Fuel Assembly Assuming All the Bond Sodium Burns

Assumptions

1. All Na in assembly is released and burns
2. There are 8 g of Na per pin (169 pins)
3. The assembly container tube is assumed to be in contact with the assembly and heat is transferred to the tube. The container tube would be crushed by any hypothetical accident that crushed the fuel assembly.
4. All heat generated remains in the assembly and container tube, and is uniformly distributed. Sodium fires tend to be relatively slow so that considerable conduction will occur in the fuel assembly metal components, which will tend to produce uniform temperatures. In fact significant conduction to the container and cask liner, which is neglected, will occur.
4. The assembly consists of metal fuel with stainless steel cladding and duct with an initial temperature of 427 °C (800 °F). The specific heat for HT-9 is higher, so it conservative to assume stainless steel (2).
5. The fuel rod diameter is 0.498 cm (0.196 inch) and the total fuel plus blanket length is 122 cm (4 ft).

Material Properties	U-metal(1)	316(1)	Na(3)
Specific Heat, (J/g·C)	0.12	0.52	---
Density (g/cm ³)	15.8	7.8	---
Heat of Combustion (J/g)	---	---	9500

Mass of a fuel assembly is approximately 400 lb (181 kg)

Fuel mass = 169 pins x 15.8 g/cm³ π (0.498cm)²/4 x 122 cm/1000 g/kg

Fuel mass = 63 kg

Assembly steel mass = 181 kg - 63 kg = 118 kg

Tube container mass of upper portion of tube is 17 cm (6.69 inch) in diameter, with a wall thickness of 0.277 cm (0.109 inches). Length is 335 cm (11 ft.)

Tube mass = 7.8 g/cm³ π [(17cm)² - (16.45 cm)²]/4 x 335 cm/1000 g/kg = 37.8 kg

Heat generated = mass of sodium times heat of combustion
= 169 pin x 8 g/pin x 9500 J/g = 1.28 x 10⁷ J

$$\text{Temperature Increase} = \frac{0}{[m(\text{fuel})c_p(\text{fuel}) + c_p(\text{steel})(m_{\text{steel}}(\text{assembly}) + m_{\text{steel}}(\text{tube}))]}$$

$$= \frac{1.28 \times 10^7 \text{ J}}{(6.3 \times 10^6 \text{ g} \times 0.12 \text{ J/g} \cdot ^\circ\text{C} + 0.52 \text{ J/g} \cdot ^\circ\text{C} \times (1.18 \times 10^5 \text{ g} + 3.78 \times 10^4 \text{ g}))}$$

$$= 145 ^\circ\text{C}$$

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PEER REVIEW CHECKLIST

Document Reviewed: Effects on Dose Consequences of Storing Metal Fuel in the
Interim Storage Casks

Author: J. C. Van Keuren

Date: January 20, 1997

Scope of Review: Entire Document

Yes No NA

☐ ☐ ☒

Previous reviews complete and cover analysis, up to scope of this review, with no gaps.

☒ ☐ ☐

Problem completely defined.

☒ ☐ ☒

Accident scenarios developed in a clear and logical manner.

☒ ☐ ☒

Necessary assumptions explicitly stated and supported.

☒ ☐ ☒

Computer codes and data files documented.

☒ ☐ ☒

Data used in calculations explicitly stated in document.

☒ ☐ ☒

Data checked for consistency with original source information as applicable.

☒ ☐ ☐

Mathematical derivations checked including dimensional consistency of results.

☒ ☐ ☐

Models appropriate and used within range of validity or use outside range of established validity justified.

☒ ☐ ☐

Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.

☐ ☐ ☒

Software input correct and consistent with document reviewed.

☐ ☐ ☒

Software output consistent with input and with results reported in document reviewed.

☐ ☐ ☒

Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.

☒ ☐ ☐

Safety margins consistent with good engineering practices.

☒ ☐ ☐

Conclusions consistent with analytical results and applicable limits.

☒ ☐ ☐

Results and conclusions address all points required in the problem statement.

☐ ☐ ☒

Format consistent with appropriate NRC Regulatory Guide or other standards

☐ ☐ ☒

Review calculations, comments, and/or notes are attached.

☒ ☐ ☐

Document approved.

D.A. Himes

1/22/97

Reviewer (Printed Name and Signature)

Date