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FFTF DESIGN FLEXIBILITY
FOR
HYPOTHETICAL ACCIDENT MITIGATION

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UNITED STATES ATOMIC ENERGY COMMISSION

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R.E. Peterson

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FFTF DESIGN FLEXIBILITY FOR HYPOTHETICAL ACCIDENT MITIGATION

INTRODUCTION

The capability of the Fast Flux Test Facility (FFTF) containment to withstand consequences associated with a hypothetical core disruptive accident (HCDA) has been investigated over a wide range of energy release. Responses of the primary coolant system boundary, the reactor cavity and the containment building to a theoretically available work energy of 150 MW-seconds have been evaluated. The sensitivity of the design to work energy was also assessed by extending the calculations to 350 MW-seconds.⁽¹⁾ More recently, in response to regulatory questions, the design has been evaluated for yet arbitrarily higher energy releases up to 1000 MW-seconds. In these evaluations, the results of which were submitted to the Division of Reactor Licensing (DRL) previously,⁽²⁾ responses of the following design features were analytically obtained:

- Reactor vessel head
- Reactor cavity pressures with and without vapor bubble collapse
- Piping interferences
- Plugs in the reactor vessel head
- Head compartment pressure sensitivity to postulated sodium release
- Reactor and guard vessel
- Reactor support system

Conclusions from these evaluations are given in the following brief summary.

(1) D. D. Stepnewski, G. L. Fox, R. D. Peak, and K. R. Merckx, *Mechanical Consequences of Hypothetical Core Disruptive Accidents*, HEDL-TME 71-50, Hanford Engineering Development Laboratory, Richland, Washington, April 19, 1971.

(2) Letter of transmittal, M. Shaw to P. Morris, *Safety Review of FFTF (P-448)*, September 27, 1971, RDT:NS:AE:408.

The FFTF design provides the capability for containment of an energetic hypothetical core disruptive accident. The main characteristics of this containment capability include:

- Maintenance of the reactor cavity as an effective barrier for radioactivity before its escape to the containment building
- Decay heat removal from the damaged core, within the reactor vessel, through the heat transport system to the air-cooled heat dumps
- Ultimate retention of radioactivity in the containment building with restricted leakage to the environment.

The system is being designed with the full capability to sustain all three containment characteristics at a hypothetical accident energy level of 150 MW-seconds. Recent evaluation studies, together with the earlier containment evaluation, have further assessed containment characteristics, with the following conclusions:

- a. Failure modes and the extent of damage identified are not expected to prevent circulation of sodium by the main heat transport system for accident energy levels up to 350 MW-seconds. Potential impairment of individual HTS component functions in this range is possible and requires further evaluation.
- b. Effectiveness of reactor head seals required for leaktightness of the reactor cavity is difficult to assess. For energy levels above 150 MW-seconds a reduction in seal efficiency may well occur. The two major seals which must be designed to accommodate significant displacements and through which significant sodium expulsion is a consideration, i.e., the closed loops and the canopy seal around the head periphery, are believed capable of remaining intact beyond an energy level of 350 MW-seconds.
- c. Evaluation of system performance becomes more uncertain considering the energy range of 350 to 1000 MW-seconds. The assessment of containment capability includes:

- In the lower part of this energy range, major seals in the reactor head are expected to be impaired and reactor and heat transport damage would become extensive enough that continued heat removal by the heat transport system is questionable
- The reactor head is believed capable of remaining intact as an effective barrier against sodium spray expulsion for energy levels well into this range
- It is expected that structural integrity of the reactor cavity can be maintained for energy levels well into this range and perhaps beyond.

As has been pointed out previously, no realistic mechanism for the occurrence of an accident of this type in FFTF can be identified. Therefore, design provisions for limiting the consequences of an HCDA are discussed in terms of margins against unforeseen events. Thus, in addition to the capabilities which are inherent in the design, two additional features have been provided, which could provide greater flexibility in the mitigation of accident consequences should a need be identified. Their use is not presently deemed to be necessary. These features are:

- Space below the reactor cavity to accommodate an ex-vessel post-accident heat removal system
- Ability to include a sealed deck over the head compartment to provide an additional barrier against releases to the containment vessel interior.

The above features would not be implemented unless subsequent developmental results should yield evidence that the progression of accident consequences departed significantly from those presently envisioned. For example, if in-vessel post-accident heat removal could not be assured, consideration might be given to an alternate post-accident heat removal approach using the space below the cavity. It should be noted however that recent evidence from experiment and from analysis is establishing that in-vessel post-accident cooling is adequate under the conditions postulated.

It is the purpose of this report to document the information previously provided on these additional features and to summarize the flexibility afforded in the plant as it is proposed to be constructed. Further detail is

also included to establish that implementation of these features is feasible if needed to meet established siting criteria.⁽¹⁾ Specifically, for the compartment below the reactor cavity, the access provisions, design criteria for the reactor cavity floor, and placement of support columns are discussed. For the head compartment area, details of the deck and deck anchorage are included.

Post-Accident Heat Removal

To better illustrate the flexibility afforded by the space below the reactor cavity it is desirable to first describe the phenomena involved in post-accident heat removal. By way of review, the approach for cooling a severely disrupted core following a hypothesized energetic power excursion in FFTF is as follows:

- The primary heat transport system is calculated to retain its cooling capability -- thereby ensuring the ability to transport decay heat from the vessel to the heat dumps
- The core support structure provides a stable bed to support core debris and to distribute coolant flow up through debris and out to the heat transport system
- Debris continues to be cooled within the vessel for an indefinite period of time.

The characteristics of core debris, the lines of defense available to maintain in-vessel cooling and design provisions for instituting yet another line of defense beneath the reactor cavity were previously summarized.⁽²⁾ The information presented in this report further elaborates on these items.

A. Core Retention and Cooling

Factors considered for post-accident heat removal in FFTF include the characteristics of core debris immediately following a disruptive accident, mechanical integrity of the primary system boundary, and thermal hydraulic limitations governing heat removal. Core debris characteristics of interest are debris distribution, decay heat energy, and chemical and physical characteristics.

(1) *Fast Flux Test Facility Preliminary Safety Analysis Report, Appendix F, Principal Design Criteria*, September, 1970.

(2) Letter of transmittal, M. Shaw to P. Morris, *Safety Review of FFTF (P-448)*, September 27, 1971, RDT:NS:AE:408.

Evaluations completed for FFTF on both the mechanical integrity of the primary system and core debris characteristics provide a technical basis for the approach of providing post-accident heat removal using the primary heat removal system. This approach is based on initial analytical and ultimate experimental substantiation that mechanical integrity of the primary cooling system sufficient for decay heat removal is preserved after the energy release. Heat removal from the debris was shown to be sufficient to prevent melting penetration of the core support structure and the reactor vessel.⁽¹⁾ The basic heat transfer phenomena are being verified experimentally by simulated debris heat transfer tests in sodium. It should be noted that these thermal considerations are essentially the same for ex-vessel approaches to post-accident heat removal.

B. Core Debris Characteristics

In arriving at the bases for cooling of core debris the first questions considered were those relating to the characteristics of core debris immediately following the accident.

First, only a relatively small fraction of fuel will lose its basic structural integrity -- most will be retained in the fuel assembly. Any vaporized or molten core material released to the coolant channels will be fragmented. Experimental data substantiate this fragmentation and also yield direct measurement of debris particle size distribution to be used in heat transfer analysis. It is the fragmentation phenomenon that causes formation of porous debris that can be effectively cooled. The balance of the core is expected to be distorted but retain its rod geometry. Although local melting may occur due to this distortion the process is self-quenching because of the fragmentation which accompanies it, the high heat transfer coefficient, good conductivity and the high boiling point of sodium are factors favorable to promoting heat transfer even if substantial core rearrangement takes place.

The amount of heat that must be removed from debris beds is another factor in establishing in-vessel retention. Fuel that melts will release gaseous

(1) *Post-Accident Heat Removal in LMFBF's, In-Vessel Considerations*, ANL/RAS 71-12, Argonne National Laboratory, Argonne, Illinois, March 1971.

and volatile fission products rapidly so that the heat remaining with debris that contacts supporting structures is reduced initially by nearly a half. It is possible that certain volatiles may react during the HCDA with fuel, cladding, and structure to form nonvolatiles or be retained within the cooled fuel matrix. The heating rate from debris without gaseous or volatile fission products is about 1 cal/sec-gm after the first minute and after the first day this drops to about 0.15 cal/sec-gm, or 0.63 watts/gm of fuel. Considering the total heat that must be removed from the vessel if it includes all particulate debris, remaining fuel and escaped fission products, this amounts to 5-1/2 percent and 0.5 percent of full core power at the end of one minute and one day, respectively. Natural convection in the heat transport system is more than adequate to remove this heat.

C. In-Vessel Retention

Having examined the characteristics of core debris following such a severe accident, the feasibility of in-vessel retention from both a mechanical and a thermal point of view was considered in detail.

From a mechanical viewpoint potential damage to the primary system boundary consisting of the vessel, primary piping and components was considered first. As previously stated, the primary system is being designed to accommodate work energy releases of 150 MW-seconds. Evaluations of the capability at an energy level of 350 MW-seconds are being conducted as the design progresses.

Dynamic analyses have also been carried out for the core support structure. In the following evaluation reference is made to Figure 1 in discussing the regions of maximum stress. Dynamic loading calculations were performed using a NASTRAN structural model and pressure-time histories for 350 MW-seconds developed from REXCO and PISCES codes.^(1,2,3) Using a 15-degree

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- (1) Caleb W. McCormick, *The NASTRAN Users Manual*, NASA SP-222 (1969).
 - (2) Y. W. Chang, J. Guildys, and S. H. Fistedis, *Two-Dimensional Hydrodynamic Analysis for Primary Containment*, ANL-7498, Argonne National Laboratory, Argonne, Illinois, November 1969.
 - (3) *An Introduction to the PISCES TM System of Continuum Mechanics Codes*, Physics International Company, San Leandro, California, September 1970.

Figure 1. Core Support Structure

sector model, stress histories were obtained for all elements including the skirt area where maximum stresses of significance are anticipated. While further analyses are proceeding in this area with improved dynamic models of the core support structure, preliminary stress values in the skirt area show adequate margins against failure.

Heat transfer from core debris within the vessel is considered in each of five specific areas:

- In the core
- On top of the core support structure between vessel and core barrel
- On the bottom plate of the core support structure
- On the bottom of the vessel
- On the bottom of the guard tank

The characteristics of the fuel-sodium system result in porous beds in each case with decay heat being given up to the sodium pool such that melting penetration is prevented. Recent experimental data from Argonne National Laboratory appears to substantiate earlier estimates of heat transfer from such flooded beds.

The ANL experiments consisted of beds of fragmented UO_2 flooded with sodium and heated from beneath as shown in Figure 2. Temperature measurements were taken at the bottom of the bed and in the pool as a function of time under boiling conditions. Bed density was determined radiographically after the bed had cooled. Results from these tests are shown in Table 1. Using a fixed bed model for calculating maximum heat flux, it was found that the experimentally determined heat fluxes were greater than calculated. This fact correlates with observed bed porosities which were higher than expected, enabling greater heat removal. Dryout in most cases was beyond the heat source capability. It is worthy of note that although dryout was finally achieved on the lower surface of beds about three inches in depth, the realistic situation in which bed heating is internal should result in even better heat transfer. It is inferred from the experiments to date that thermal burnthrough of core debris resting on the core support structure

Figure 2. ANL Porous Bed Conductivity Tests

TABLE 1
SUMMARY OF PAHR BOTTOM HEATING EXPERIMENTS

Exp. No.	Particle Size Range (μ)	Wt. UO ₂ G	Grams ₂ Per Cm ² UO ₂	ρ In Na g/Cm ³	Bed Depth (In.)	t _B °C	Peak Pyro °C	Q Btu/hr Ft ²	Q Cal/Sec Cm ²
4	Na only	--	--	--	--	907	991	175,000	--
5	25-100	310	4.35	1.92	1.5	907	1026	178,000	13.3
6	25-100	619	8.68	1.92	3.0	907	1065	210,000	15.8
8	100-300	1001	14.04	3.69	1.5	923	1079	143,000	10.7
9	100-300	1495	20.98	3.30	2.5	908	1055	107,000	8.06
+10	300 600	1413	19.83	(4.16) 5.20	1.5	935	1107	197,000	14.8
+13	600-1000	1495	20.98	4.41	1-7/8	915	1120	108,000	8.1
+17	600-1000	2518	35.34	4.64	3.0	909	1447	39,000	--
+18	600-1000	2518	35.34	4.64	3.0	926	--	45,000	--
+19	600-1000	2518	35.34	4.64	3.0	932	1384	42,000	--
+20	Na only	--	--	--	--	934	1399	37,000	--
+22	Na only	--	--	--	--	904	1350	161,000	--
23	100 1000	701	9.83	4.39 (3.16)	1.0	908	1098	190,000	14.1
24	100 1000	1410	19.7	(4.65) 4.70	2.0	904	1180	185,000	13.7
26	100 1000	3153	44.0	5.23	3.6	912 -1018	1188	23,000**	1.7
31	100 1000	2208	30.9	4.58	3.0	880 -1008	1206	67,700	5.0
32	100 1000	2208	30.9	4.58	3.0	870 -1012	1416	108,000*	7.9

† With 200 μ opening screen on bottom

* Burnout

** Heated to burnout

is highly unlikely. Further experiments and analyses are expected to provide additional confirmation.

D. Ex-Vessel Retention

Space accommodation and access provisions are incorporated in the design to permit an ex-vessel core retention system, should a valid need for such a system be identified. This reserved space beneath the reactor cavity is shown in Figure 5, page 16. Approximate dimensions are 21-ft diameter and 11-ft height within the cavity. The volume is approximately 3800 ft³. Access to the cavity for construction and pipe routing from the adjacent equipment space is provided. Further details of this feature and the design of the reactor cavity floor which supports the reactor guard vessel are discussed later in this report.

Two concepts were considered for ex-vessel core retention systems:

1. A catch grid submerged in a sodium pool, and
2. A sacrificial bed of ceramic material.

Because the first concept involves essentially the same principles as demonstrating that the debris remains in the reactor vessel in the first place, earlier efforts were directed to establishing the feasibility of the sacrificial bed concept.

The principle used is one in which molten fuel, assumed to result from complete loss of cooling, ultimately comes to rest on a bed of silica-containing ceramic material. Nominally, we have assumed the material to be basalt, which consists primarily of silicon, calcium and aluminum oxide and is relatively free of volatile materials and water. Another material that appears to perform as well or better is Olivine, an iron-magnesium silicate. As the UO₂ meltfront penetrates the ceramic a dissolution process takes place as shown on the solubility curve in Figure 3. As dissolution progresses, successive layers of liquid and solid phases exist within the meltfront as the temperature varies down to the point where the basalt is in equilibrium with the liquid. As more basalt is melted, the pool enlarges and the temperature decreases as heat flows into heat sinks surrounding the sacrificial bed in the cavity. The heat sinks presently envisioned would require auxiliary cooling of some type.

Figure 3. Estimated Solubility of UO_2 in Molten Basalt

Pool temperature as a function of radius is shown in Figure 4 for two materials of interest. At a limiting radius of 11 feet, as dictated by space considerations the meltfront ceases since it approaches a thermal buffer of fire brick refractory of higher melting temperatures. This occurs in from three to four days, depending on the material. It is evident from the curves that the sensitivity of the temperature-to-pool dimensions is not large beyond about 8 or 9 feet. It is thus concluded that space dimensions are probably adequate but substantial development efforts are required to confirm the feasibility of such a concept.

Results of static solubility measurements at temperatures up to 2950°F were indicative of the desired behavior. Viscosity measurements for basalt that are used in analysis of convective heat transfer were obtained. Reducing the viscosity to improve convective heat transfer using additives such as B_2O_3 is an option. Some testing of commercial refractories at the anticipated interface temperature of the molten pool was also done. MgO in contact with molten basalt appeared to have acceptable corrosion rates.

The results obtained in these preliminary tests are not sufficient to resolve serious feasibility questions. If development of the concept to a fully substantiated status were to be undertaken, much more work of this type would be required.

The catch grid-sodium pool concept has been considered in several fast breeder reactor design studies. A review of this concept indicates that the tank and catch-grid could be fitted within the cavity space provided. This concept operates on the principle of natural circulation of sodium within a tank to cool core debris which settles on the grid contained within the tank. Heat must be removed from the bulk sodium by some means such as cooling coils submerged in the sodium within the tank. The cooling coils, in turn, would be connected to a separate heat removal system similar to that required for the heat sink in the sacrificial bed concept described above. This system could be tied into the secondary side of the main heat transport system. Another alternate possibility is the cell cooling system. The technology upon which this system is based is very similar to the in-vessel reference approach.

Figure 4. Pool Temperature and Penetration Radius

Significant design and development work would be required with either of the two concepts described, if an ex-vessel system was to be adopted. An assessment of both concepts indicates that space and access are adequate to accommodate either with reasonable confidence.

Reactor Cavity Design

The reactor cavity design, including the lower cavity is shown in Figure 5. Detailed design is in progress to the following requirements:

If use were to be made of the lower cavity space, it would be prior to initial startup of the reactor; therefore, there is no shielding requirement on the reactor cavity floor which separates the reactor cavity from the cavity below. Walls of the cavity below should have shielding equal to the reactor cavity. Maintenance or inspection access to the lower cavity, if required, will be provided within the two blockout openings specified for construction purposes.

The floor of the reactor cavity is to be designed for the 35 psig in the reactor cavity and to separate the reactor cavity from the cavity below for all sodium spills. The reactor cavity floor is also to be designed for reactor guard vessel support loads under accident assumptions. The assumed load corresponds to deformation of the bottom of the guard vessel in the event of interference with the reactor vessel. A 4-inch interference results in a maximum of 6×10^6 pounds being transferred to the reactor cavity floor by the guard vessel support.

In designing the reactor cavity floor, consideration is being given to a design which allows melt-through of the center portion of the floor, but does not destroy the support of the guard vessel. Structural steel is considered more appropriate than concrete for the reactor cavity floor because of the hydrogen release from molten fuel in contact with the concrete.

The lower cavity is an extension of the reactor cavity only upon melt-through of molten fuel from the reactor cavity above. Walls and floors of the lower cavity are to be designed for 35 psig internal pressure. Concrete design temperatures for the lower cavity are the same as the reactor cavity.

Figure 5. Reactor Cavity Concept

Embedded members are planned for inclusion in the lower cavity walls to facilitate installation of a steel cell liner at a future date or for possible attachment of equipment or coolant lines. If a cell liner is provided later, it would be continuous with the reactor cavity liner so that upon melt-through there will be minimum leakage from the two cavities.

Two construction openings are also planned for access to the lower cavity. These are visualized as approximately 6-ft diameter blockouts on opposite sides of the cavity and at the upper elevation of the cavity. These would be used for construction access and then filled in except for coolant lines and maintenance access plugs that may be required if equipment were to be installed.

Six steel support columns are placed at 60-degree intervals around periphery of the reactor cavity. These columns provide structural support of the ring forging and building structure from which the reactor vessel is supported. These columns are anchored into the lower cavity concrete and will accommodate either up or down reaction forces that may be imparted to the vessel by hypothetical accidents that have been considered.

FFTF Head Compartment

The FFTF head compartment is that area above the reactor vessel head bounded by the operating deck on top and the compartment walls on the sides as shown on Figure 6. The head compartment provides:

- Space for equipment in the top head area
- Access for maintenance, refueling, and other operations
- A suitable environment for safe manned access
- Radiation and safety protection for personnel in the operating floor area as well as the head area.

The main equipment items located in the head compartment are:

- Three IVHM mechanisms and plugs
- Three instrument tree mechanisms and plugs
- One centrally located complex of control rod drive mechanisms and closed loop tubes surrounded by shielding and located directly over the reactor core
- Three pathways of branch arm piping and associated shielding.

Figure 6. FFTF Head Compartment

The branch arm piping is that special section of piping within the head compartment which connects the vertical closed test loop pressure tubes to the rest of the piping running to the closed loop cells. The piping carries the sodium coolant which is used to provide the closed loop test environment.

Ready access to the head compartment for operations, maintenance, and inspection during day-to-day operations is most important and as such the compartment is being designed to permit manned access and maintenance during operation without the use of protective equipment or clothing. In this regard, an open grating design is actively being investigated to be used as the reference design to satisfy these requirements. The capability of the reference open grating design to deflect sodium and provide missile protection consistent with the appropriate requirements to protect containment integrity will be retained.

However, as previously indicated, flexibility in the design is also being retained to include a backup design.⁽¹⁾ This backup design would be a solid carbon steel structure that covers the entire head compartment area. Ports that would be provided for personnel access to the head compartment and atmospheric communication between the head compartment and the containment would be limited to approximately 300 ft² of louvered vent area around the periphery of the deck. This design would allow even this small vent area to be sealed so as to completely isolate the head compartment atmosphere during normal reactor operation. Embedded steel will be installed in the walls of the compartment so as to be able to anchor the head compartment deck for up to 35 psig should it be decided at a later date to maintain a tight compartment for the backup design. Figure 7 shows the pressure deck anchorage detail that will be provided.

(1) Letter of transmittal, M. Shaw to P. Morris, *Safety Review of FFTF (P-448)*, September 27, 1971, RDT:NS:AE:408

Figure 7. Deck Anchor Detail