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ACCIDENT ANALYSIS FOR U.S. FAST BURST REACTORS

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ABSTRACT

In the U.S. fast burst reactor (FBR) community there has been increasing emphasis and scrutiny on safety analysis and understanding of possible accident scenarios. This paper summarizes recent work in these areas that is going on at the different U.S. FBR sites. At this time, all of the FBR facilities have or in the process of updating and refining their accident analyses. This effort is driven by two objectives: to obtain a more realistic scenario for emergency response procedures and contingency plans, and to determine compliance with changing regulatory standards.

I. INTRODUCTION

In the U.S., the operation of fast burst reactor facilities (Table 1) is documented in a safety analysis report (SAR). The SAR is a comprehensive document describing the facility site, the reactor(s) and control system(s), support operations, and the risk associated with FBR operation. The SAR forms the basis for several other facility documents, including emergency evacuation plans and the technical safety requirements (TSR), formerly called the technical specifications.

One of the most important aspects of the SAR is the accident analysis. The accident analysis describes the credible accident events and the potential consequences. This accident analysis forms an envelope of permissible operations which may be conducted at the site. The accident analysis must include a qualitative evaluation of risk and a quantitative analysis of the technical consequences to the FBR assembly, the facility, site personnel, and the public-at-large.

Table 1 - Operating U.S. FBR Facilities

FBR	Site Location	Reactor Housing	Nominal Pulse Fissions
APRF	Aberdeen Proving Gnd.	Metal silo	2×10^{17}
Godiva-IV	LANL, Los Alamos	Concrete bldg.	5×10^{16}
Molly-G	WSMR, White Sands	Underground cell	1×10^{17}
Skua	LANL, Los Alamos	Concrete bldg.	2×10^{17}
SPR-II	SNLA, Albuquerque	Concrete bldg.	1×10^{17}
SPR-III	SNLA, Albuquerque	Concrete bldg.	2×10^{17}

Safety analysis has concentrated on two classes of accidents:

1. the minor operational event or "mishap" in which little or no activity is released, and
2. the maximum-consequence accident or maximum credible accident (MCA), which postulates the worst-case scenario endangering the public-at-large.

The first class of accidents includes loss of site power, reactivity insertions from a single runaway control rod, natural disasters such as fire, flooding, and earthquakes. For FBR facilities, the consequences of these accidents are relatively minor and can be readily controlled within the confines of the facility.

The second class, the MCA, is generally not a credible event, but a very conservative assumption which can be used to establish the facility design and operating parameters, such as shielding thickness and restricted area boundaries. This type of postulated accident is also called the Design Basis Accident (DBA). The MCA is usually defined in terms of the total number of fissions produced. The results are damage to the FBR, including local melting and a small amount of fuel vaporization, with subsequent release of radionuclides to the facility and the local site. The event postulates insertion of the excess available reactivity without specifying the event initiator. These extremely unlikely events ($< 10^{-6}$ per year probability) nevertheless define an envelope of experiments and operations possible at the facility. In addition, they assist in specifying the maximum off-site dose commitments and emergency preparation measures.

II. OPERATION OF FAST BURST REACTORS

Fast burst reactors are specifically designed to produce precise, reproducible, self-terminating, prompt critical bursts of neutron and gamma radiation. The first burst assembly, Lady Godiva,[1,2] began operation in 1951 and was originally intended as a bare, spherical critical mass of U(93) (uranium metal of 93.2 wt% ^{235}U). The unreflected assembly was basically a 6.8-in.-dia. sphere in three horizontally parted segments weighing a total of ~53 kg.

The original Godiva was replaced by Godiva-II[3] in 1957. The core geometry was a nearly cylindrical version of Lady Godiva, weighing ~57.7 kg. It was, however, specifically designed for burst production. Godiva-II was the forerunner of a generation of burst assemblies built at Sandia, Oak Ridge, White Sands, and Aberdeen Proving Ground. The latest version, Godiva-IV,[4] is primarily an irradiation facility; although its original purpose was to test design features, including material selection, that are expected to increase resistance to shock damage. Characteristics of several U.S. fast burst assemblies are shown in Table 2.

Figure 1 shows the two major U(93)-alloy parts of Godiva-IV, a stationary head and movable safety block, which form an essentially unreflected cylinder when brought together remotely. In this condition, delayed criticality can be attained by adjustment of two U(93) control rods that enter the head from below. From this state, a burst may be produced, allowing a further slight adjustment of control-rod position, by a sudden reactivity addition brought about by insertion of an interlocked U(93) burst rod. Step reactivity insertions are possible due to the low intrinsic neutron source strength of U(93). Thermal expansion terminates the burst; the associated shock ejects the safety block.

The production of a burst of known magnitude involves a well-defined cycle. This cycle includes

- a delayed critical check,
- a control adjustment to trim excess reactivity as required for the desired burst while allowing for temperature drift,
- the retraction of the safety-block to allow decay of the delayed neutron population,
- the reinsertion of the safety block, and
- burst-rod insertion.

Interlocks prevent major departures from this cycle. The burst actuates a scram signal, which deactivates a magnet that normally secures the safety block and ejects the burst rod. Variations on this theme have evolved over the years.

Table 2 - Core Characteristics of Several FBRs

FBR	Fuel Material	Core Configuration	Safety Block Configuration	Control Elements Configuration
APRF	U(93)-10% Mo	Bare Solid Cyl.	Central Cyl.	2 Vernier + 1 Burst Rod
Godiva-IV	U(93)-1.5% Mo	Bare Solid Cyl.	Central Cyl.	2 Vernier + 1 Burst Rod
Molly-G	U(93)-10% Mo	Bare Solid Cyl.	Central Cyl.	2 Vernier + 1 Burst Rod
Skua	U(93)-1.5% Mo	Refl. Annular Cyl.	3 Radial Refl.	2 Vernier + 1 Burst Drum
SPR-II	U(93)-10% Mo	Bare Solid Cyl.	Split Core	2 Vernier + 1 Burst Rod
SPR-III	U(93)-10% Mo	Refl. Annular Cyl.	Split Core	3 Vernier + 1 Burst Refl.

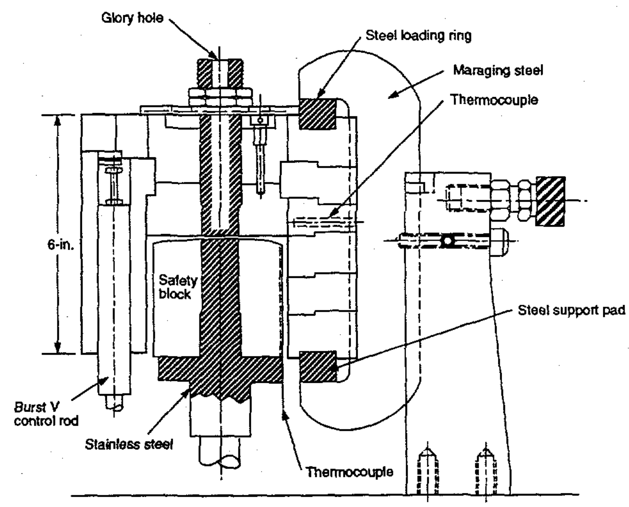

Fig. 1. Godiva-IV fast burst assembly machine.

III. DEFINITION OF FBR MCAS

It is the ability to perform step prompt reactivity insertions combined with the fact that protect systems cannot act fast enough to affect the burst energy yield that characterize the MCAs for FBRs. FBRs must operate within a narrow reactivity envelope between prompt critical and typically 0.10 $ above prompt. For these reasons FBRs are subject to strict procedural and administrative control.

FBR MCAs are not based on delineation of physical mechanisms, but rather on the physical limits of reactivity insertion. The excess reactivity is limited by the design of the control elements and the control system. The reactivity control elements and the control instrumentation are specifically designed to minimize errors in execution and control element positioning. The physical mechanisms necessary to initiate these accidents require multiple simultaneous operator errors, failures of control interlocks, and breakdown of administrative controls. Deliberate actions are excluded from consideration since they are treated under security procedures associated with FBR operations.

The excess prompt reactivity inserted prior to a burst determines the energy release or yield. The free-field prompt excess reactivity for U.S. FBRs is shown in Table 3. FBR MCAs assume insertion of excess reactivity without regard to preinitiation. Preinitiation limits the yield in a reactivity-induced excursion by inducing the chain reaction before all reactivity elements have been fully assembled. The probability of obtaining the full yield for a 0.40 $ prompt insertion with the intrinsic neutron

Table 3 - MCA Specifications for FBR Sites

Facility	MCA Prompt Reactivity	MCA Yield Fission
APRF	0.25	3×10^{18}
Godiva-IV	0.40	1.7×10^{18}
Molly-G	0.25	1×10^{18}
Skua	-	-
SPR-II	0.40	3×10^{18}
SPR-III	-	4.1×10^{18}

source in Godiva-IV is ~ 57 %.

MCA accidents do not produce an explosive disassembly, but the fission density is sufficient to produce local melting in the core hot spot. However, little or no fuel vaporization occurs. In the APRF accident an energy release of 6.1×10^{17} fissions (18 MJ) caused partial melting (~10%) of the safety block and cracking of the central fuel rings. The event rendered the machine unusable and contaminated, but the confinement building remained intact.

These events define an envelope for experiments at FBR facilities because they generate the maximum consequences given the scope of operations that can take place. For example, the effect of a proposed experiment and its particular accident potentials can be assessed (in terms of the consequences of the accidents already analyzed in the facility SAR) through a qualitative analogy related to system safety. These events also define the level of emergency evacuation procedures necessary in the event of a given accident.

IV. REGULATORY GUIDANCE FOR MCA ANALYSIS

FBR operations are characterized by high power bursts (50,000 to 100,000 MW peak power) for short duration (30 to 300 μs) with typically one to three operations conducted daily. This results in a low fission-product inventory and low average core activity in comparison to other types of research reactor facilities. The graded approach toward risk management allows FBRs to operate without the same degree of regulatory scrutiny afforded to power reactors or high-power research reactors. Nevertheless, the accident analysis must consider the risk to the site personnel and the public.

Guidance for research reactor SARs is forthcoming in ANS draft standard 15.21.[5] This standard specifies information and analysis for inclusion in research reactor SARs and establishes a standard format. The assessment of the consequences of a postulated radionuclide release which bounds all other events is discussed, and a deterministic accident analysis approach is recommended. It is further recommended that the detail of the analysis depend on the potential risk caused by the operation.

ANS/ANSI Standard 15.7 provides guidance for defining the boundary zones surrounding FBR sites in terms of the time necessary to achieve evacuation of the population.[6] It provides additional guidance on dose commitments within the boundary zones. Table 4 gives the recommended maximum dose commitments at the various boundary zones.

For DOE-licensed operations, DOE Order 6430.1A[7] specifies radiological siting requirements and siting guidelines that define the limits of operations that can be conducted at U.S. facilities. It requires that "events to be analyzed are those judged to have maximum consequences" and that "radiation dose to an off-site individual receiving maximum exposure shall be evaluated." Furthermore, the order requires that "dose assessment shall consider both the duration of the event and, consistent with emergency response capability to control or evacuate individuals, the duration of exposure."

V. ANALYSIS OF FBR MCAS

FBR MCAs are hypothetical accidents based not on identification of precise physical mechanisms but rather on the limits of available reactivity insertion. Given the scope of FBR operations, they lead to maximum consequences. The scenarios and the analytic methodology are extremely conservative. These postulated scenarios lead to significant calculated doses inside the FBR facility boundary, but, in most cases, significantly less than radiological siting guidelines of ANSI/ANS 15.7. In any event, the dose consequences are "maximum exposure" and are not the bases for establishing operational exposure limits for FBR facilities.

Accident modeling is done in stages which correspond to the physical event (Figure 2). The analysis is preceded by evaluations of experiments for the current and future anticipated range of FBR operations that could give rise to the most serious consequences. This was followed

Table 4 - ANS/ANSI 15.7 Dose Commitments Specifications

Boundary	Whole-Body Dose (rem)	Max. Organ Dose (rem)	Time Duration (hours)
Operations Boundary	25	75	<1
Site Boundary	5	15	2
Rural Zone	0.5	1.5	2
Urban Boundary	0.5	1.5	24

by examination of the maximum excess reactivity and mechanisms for reactivity insertion.

Neutronic-hydrodynamic calculations or multi-region point kinetics are used to determine the energy release from the transient assuming full reactivity insertion and no operating protect/safety systems. The energy release from the MCA event is used to generate the source term from fission product yield data and accepted release fractions. Modeling the source release is done through release curves calculated to define the amount of material released per unit time from the point of release. Atmospheric dispersion of the released inventory uses a standard atmospheric diffusion model.

VI. RECENT EFFORTS IN MCA ANALYSIS

Work on new SARs at Los Alamos, Aberdeen, Sandia, and White Sands has drawn attention to FBR accident analysis. The Los Alamos Critical Experiments Facility (LACEF) has developed accident analysis tools for a wide variety of solution and metal critical assemblies,[8] including FBRs. The calculation of FBR MCA energy release is a complex problem. The physical phenomena range from events which produce small temperature rises to solid-liquid-vapor phase changes through explosive disassembly scenarios. Research at Los Alamos is exploring the use of coupled hydrodynamic-neutronic calculations in both one- and three-dimensions. This work is reported in another paper in these proceedings.[9]

Efforts at Aberdeen have centered on studying the release of fission product isotopes from the fuel during routine operations.[10] Air samples were drawn from the reactor air filtration and examined using a gamma spectrometer to identify fission product isotopes. This is both

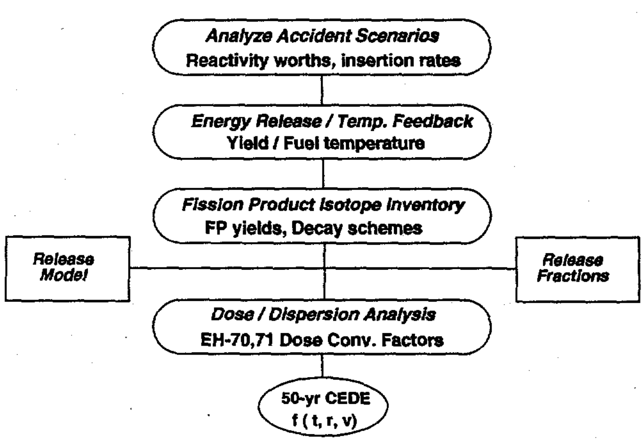

Fig. 2. MCA accident modeling.

Table 5 - History of FBR Pulse Operations

Facility	Startup Year	Number of Pulse Operations to Date
APRF	1970	4528
Godiva-IV	1968	~1700
Molly-G	1964	~6000
SPR-II		
SPR-III		

to verify compliance with air activity standard 40CFR61 and to compare with the 1968 accidental excursion. With this data it is possible to extrapolate the source release during an MCA and compare to calculated source term release models.

VII. CONCLUSIONS

FBRs are low-risk facilities in terms of their effect on the environment and any abnormal radiation exposure to the on-site personnel and the general public. This is mainly due to their inherently small fission product inventory and good radionuclide retention of the fuel alloy during normal and accident conditions. Under accident conditions any fuel melting which may occur is localized to the core hot spot (the safety block in solid-core FBRs) and is usually surrounded by solid fuel, which acts as a release barrier.

FBRs have operated successfully for many years. Table 5 lists the number of pulse operations performed without accidents or serious incidents.

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