UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 20, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON HYPOTHETICAL CORE DISRUPTIVE ACCIDENT FOR LIQUID METAL FAST BREEDER REACTORS

Dear Mr. Rowden:

In response to a request from Dr. Dixy Lee Ray in October of 1974 (later reaffirmed by the Nuclear Regulatory Commission), the ACRS has conducted a study of whether the hypothetical core disruptive accident (HCDA) should be considered as a design basis accident in evaluating the safety of the liquid metal fast breeder reactor, and to what extent provision should be made for a core retention system in the design of the LMFBR.

In the course of its deliberations the ACRS has met with representatives of the NRC, ERDA, Argonne National Laboratory, Los Alamos Scientific Laboratory, Hanford Engineering Development Laboratory, General Electric Company, and Atomics International. Several members of the Committee also visited fast reactor facilities in the Federal Republic of Germany, in the United Kingdom, and in France.

Experience in the operation of liquid metal cooled fast reactors has extended over 28 years and has included several different designs. Experience in the United States includes that acquired in operating Clementine, EBR-I, EBR-II, Fermi, and SEFOR. The EBR-II has been in operation with a high availability at or near rated power since 1964. In the United Kingdom, the Dounreay Reactor has operated since 1959, and more recently the Prototype Fast Reactor has been operated at power. In France the Rapsodie reactor has operated since 1967 and the Phenix Reactor (a prototype plant with electric power output of about 250 MW) has been operating at rated power for about two years. The USSR has operated a 5 megawatt experimental reactor (BR-5), a 12 megawatt test reactor (BOR-60), and a dual-purpose reactor (BN-350) for production of electricity and desalination of water. Operating experience with these reactors has confirmed the ability of the designers to predict reactor behavior in those situations encountered in normal operation. Although an accident during an experiment led to melting of an EBR-I core (located at the Idaho National Engineering Laboratory), and a flow blockage in the Fermi Reactor led to partial melting of two subassemblies in neither case was a pressure pulse produced, and in no case has there been a problem which produced radiological danger to the public. Both reactors were restored to operation after these events.

The HCDA, under various names, has been postulated in the consideration of fast reactor safety by U.S. regulatory groups for 25 years. During this period a large amount of effort has gone into exploration and description of the physical phenomena and the various sequences of events that might lead to a HCDA. This study has led to a significant body of opinion that a HCDA is extremely unlikely to occur. Nevertheless, should the reactor shutdown system fail to operate when called upon at the time of the occurrence of some transient — such as a large increase in power or a loss of coolant flow — the core could overheat, fuel could melt, and pressures could be developed whose magnitude and timing might be difficult to predict quantitatively and which might be capable of exerting forces on various containment barriers.

Though the likelihood that a loss-of-coolant flow or a transient overpower might be accompanied by a failure of the shutdown system is quite remote, this combination of events has historically been postulated as the initiator of a possible pressure-driven disassembly. Consequently, there have been large-scale efforts (primarily involving calculations, but including some experiments) to describe the behavior of the core following either one of the proposed initiating events. Elaborate computer codes have been devised to describe the postulated sequence of events, such as sodium boiling and voiding, fuel and clad melting, and the subsequent relocation of fuel, that might follow the initiating event.

Another class of codes has been developed which accepts as initial conditions an extremely distorted geometry (including, in some cases completely melted fuel and cladding) and a very large and continuing insertion of positive reactivity. These codes then calculate a power transient, a pressure pulse, the kinetic energy produced, and the resultant mechanical effects on containment barriers.

The ACRS notes that considerable progress has been made in the modeling of the events that might lead to production of a power transient and disruption of the core. An increase in understanding has been developed both through studies made with the codes and through associated experimental programs carried out to elucidate particularly complex physical phenomena. However, there is a consensus that the codes describing the onset of sodium boiling and the subsequent movement of fuel give results that are at best semi-quantitative once fuel melting begins, and the description of the transition from the onset of fuel and clad melting to the core configuration described by those codes which calculate energy release, proceeds primarily by plausibility arguments.

The ACRS notes that the probability of a very rapid buildup of pressure which might lead to disruption of the containment is the product of the probability of an initiating event (such as loss-of-flow accompanied by failure-to-scram) and the probability of a subsequent increase in reactivity at a rate sufficiently large to cause a general breakup and disruption of the structure. Some design groups in the U.S. propose to insure that the first probability is sufficiently small so that the second can be ignored. A convincing demonstration that this is the case would be acceptable in principle. However, such a demonstration may require considerably more operational experience and an extensive search for initiating events associated with actual reactor designs. Even then the suggested approach seems likely to present a formidable task.

In the course of its review, the ACRS has not been able to identify any sequence of events which would demonstrably result in pressures that would be difficult to contain. Such an outcome would require that the energy level build up more rapidly than anything that would be expected to occur. Nevertheless, the existence of a positive sodium void coefficient and the presence in the reactor of several critical masses make it impossible, at least with the techniques and experience yet available, to establish with certainty that a severe excursion could not take place.

In view of this situation and the presently incomplete nature of the descriptions of actual core behavior in severe accident situations, the Committee concludes that at present consideration of the core disruptive accident must be included as a part of the safety evaluation of a liquid metal fast breeder. Protective measures against its consequences should take appropriate account of the probability that large excursions are much less likely than smaller ones, and should consider the consequences of various postulated events.

Clearly the circumstances which might conceivably lead to a core disruptive accident are likely to be accompanied by a significant measure of core melting. The relatively higher, albeit low, probability of a nonpressure-producing fuel melting accident leads to the conclusion that provisions for the containment of molten fuel should receive at least as much, if not more, emphasis than the possible occurrence of a core disruptive accident.

For the present, at least, the Committee considers it prudent that plant design include provisions for dealing with a molten mass, consisting of a significant fraction of the core, in such a way that public health and safety are not compromised.

Sincerely yours,

Dade W. Moeller

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Chairman