December 17, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: SAFETY RESEARCH FOR SODIUM-COOLED FAST REACTORS

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards has devoted increasing efforts both to review of conceptual designs for liquid-metal-cooled fast breeder reactors (LMFBR) and to considerations of fast reactor safety research. In addition to several Committee and Subcommittee meetings held with individual reactor vendors and associated utility groups, Reactor Safety Research Subcommittee meetings have been held in Washington, D. C. on December 12-13, 1968, and at Argonne, Ill., on June 4-5, 1970. In this report, the Committee will provide comments on LMFBR safety research that stem from these combined activities.

At the Subcommittee meeting held in December, 1968, representatives of Atomics International, Babcock and Wilcox, Combustion Engineering, General Electric and Westinghouse, the five industrial groups that were performing 1000 MWe LMFBR design studies each described their choice of design concept and outlined safety research and development requirements which had been identified during the design studies. Representatives of Pacific Northwest Laboratories also described safety matters arising in their conceptual studies for the Fast Flux Test Facility (FFTF).

The Subcommittee meeting held in June, 1970, concentrated on four important aspects of LMFBR safety, namely fuel failure propagation, design basis accidents, mechanical effects of nuclear accidents, and post-accident heat removal. The Subcommittee had the benefit of discussions with representatives of Atomics International, General Electric, Westinghouse, Argonne National Laboratory, Pacific Northwest Laboratory, the Division of Reactor Development and Technology (DRDT) and the Regulatory Staff. In this report, the Committee will address primarily the topics reviewed at the June, 1970 Subcommittee meeting, identifying matters which it believes warrant continuing or special attention. The recommendations which follow are intended primarily to aid in focusing the safety research program on matters significant to future, safety-related design decisions.

I. Fuel Failure Propagation. Research and development are underway on fuel failure propagation as it relates to core integrity, the ability to shut down the reactor, and the role of associated instrumentation systems. Considerable progress has been made in understanding several phenomena relevant to fuel failure propagation, such as sodium superheat and voiding and the effect of the escape of fission product gas on neighboring fuel pins. Also, analytical studies of various postulated fuel pin failure modes, and preliminary experiments on the interaction between molten fuel and sodium, provide a partial basis for evaluating the possible role of instrumentation in detecting and limiting the spread of fuel failure.

The ACRS recommends that the growing experimental and theoretical R&D effort on fuel failure propagation continue to include the following among its primary objectives:

1. The development of information either to demonstrate that significant fuel failure propagation is not a serious problem, or to determine fuel element and reactor design criteria consistent with a low probability of significant fuel failure propagation.

2. The development of information and instrumentation sufficient to assure timely detection of significant fuel failure.

3. The development of information sufficient to determine design criteria for equipment capable of timely reactor shutdown in the presence of significant fuel failure.

Among the matters which relate to the above objectives and which warrant special emphasis are the following:

a. The actual failure modes of fuel elements in service.

b. The effects on fuel failure propagation potential of continued operation of fuel elements having a loss of jacket integrity.

c. Fuel element behavior in anticipated transients as a function of irradiation.

d. The magnitude and conversion efficiency of the molten fuel-sodium interaction.

e. Sodium voiding and re-entry modes and their possible effect on failure propagation or reactivity.

f. Radiation effects on core structural members which could influence the capability to shut down the reactor.

The ACRS believes that careful consideration should be given to the modification of an existing reactor facility or construction of a new facility to enable a reasonable in-pile simulation of various potential fuel failure modes on a scale approaching that of full size fuel assemblies, including thermal, mechanical, and hydraulic interaction effects, and allowing a good test of actual instrumentation response and behavior.

II. Design Basis Accidents. There has been a considerable increase in knowledge of the Doppler effect, and more powerful methods of calculating postulated low probability reactivity accidents have been developed and applied in recent years. Nevertheless, further improvements in analytical techniques leading to greater knowledge of the nature of such reactivity accidents are needed, particularly as they may apply to larger, fast reactor cores having more positive sodium void reactivity effects and, perhaps, more complex questions of power and reactivity worth distribution.

The ACRS recommends that the safety research effort on reactivity accidents continue to include the following among its primary objectives:

1. The development of analytical techniques and information to permit an assessment of the accuracy of prediction of energy yield and conversion to mechanical work for various postulated reactivity accidents.

2. The development of analytical techniques and information to permit an adequate assessment of the effects and likelihood of potentially autocatalytic or other aggravating phenomena during postulated reactivity accidents.

Included among the matters which are relevant to these objectives are the following:

a. Experimental and analytical studies to provide a better knowledge of sodium void reactivity worths and voiding rates, including potential effects of fuel failure.

b. Effects of power flattening and the associated power density and reactivity worth gradients.

c. Modes and consequences of sodium re-entry from top or bottom on a multi-assembly basis.

d. Effects of the heterogenous nature of actual cores.

III. Mechanical Effects of Nuclear Accidents. Mechanical effects, including strong pressure waves and quasi-steady-state pressure loading, need further consideration. While improved methods of calculating pressure wave effects under conditions of symmetrical geometry have been developed, and a limited number of scaled explosion-mockup-tests have been performed for specific reactor designs, questions remain with regard to the efficiency and nature of the conversion of thermal energy to pressure and mechanical work during and after nuclear burst conditions.

In particular, the behavior of the hot fuel mass, the time scale and magnitude of transfer of heat to the sodium, the resulting sodium pressures and motion of material are all more or less uncertain, thereby introducing significant questions as to the validity of the frequently used assumption of the equivalence of a charge of high explosive to a postulated nuclear burst.

The ACRS recommends that safety research and development on mechanical effects of nuclear accidents include the following among its primary objectives:

> The development, both experimental and theoretical, of information sufficient to categorize and represent quantitatively the relevant pressure-producing phenomena during and following hypothetical reactivity excursions which would be of sufficient intensity to damage the reactor structure. Studies should include the behavior of sodium and its interaction with fuel and core structures under postulated accident conditions.

IV. Post-Accident Heat Removal. Although some analytical studies, mostly heat transfer in nature, have been performed on means of accomplishing removal of fission product decay heat following a postulated low probability reactivity excursion accident, the effort in this area to date has been modest in extent, and has not included an adequate engineering study or experimental program. Since some special means may be required to maintain containment integrity following a postulated accident which leads to melting of a significant portion of the core and disrupts the core geometry, whether or not a large reactivity excursion has been involved, the study of means to accomplish post-accident heat removal represents an important element of safety research.

The ACRS recommends that the safety research and development on post-accident heat removal be expanded and accelerated and that the following be included among the primary objectives of this work:

> The development of experimental and analytical information sufficient to permit design, as appropriate, of such an engineered safety feature for an LMFBR reactor with the objective of attaining a level of confidence in its capability equivalent to that for other engineered safety features.

Among the matters which relate to the above objective are the following:

a. The role of the various fission product species in producing decay heat throughout the primary system.

b. The potential magnitude and effect of molten fuel-sodium interactions.

c. Short and long term compatibility prgblems between hot fuel and retention structure.

The ACRS recommends that careful consideration be given to the use of in-pile experiments to aid in accomplishing this objective.

Considerable emphasis has been placed on the use of FFTF design and safety analyses to assign priorities and to choose reactor conditions for study in the LMFBR safety research program. The ACRS supports use of an actual reactor design project to crystallize thinking in safety research and development and to help identify the most pressing problems. The Committee also wishes to emphasize the importance of pursuing an overall LMFBR safety research program which anticipates and meets in timely fashion the safety research needs of forthcoming demonstration size and larger LMFBR reactors which may be proposed for typical power reactor sites.

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman