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SIMULATION OF LMFBR*

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Introduction

The title of this session is taken to imply the system-wide thermohydraulic simulation of liquid metal fast breeder reactors (LMFBR). One is interested in predicting the temperatures, pressures, and the coolant flow rates throughout the entire plant including the reactor core, the primary and secondary sodium heat transport circuits, the steam generating system as well as other auxiliary circuits. Such a simulation is needed for 1) scoping studies (i.e., in the pre-design phase of a plant), 2) detailed design development, 3) the safety analysis (post-design development phase), and 4) the operator training and plant operation. The required degree of sophistication will, of course, be different for these phases. For example, the design of structural components in the upper plenum of the reactor tank may require accurate characterization of the temperature field in the three spatial and temporal space, while, from the operating point of view this level of sophistication is not needed. It is, therefore, evident that a flexible simulation approach is essential.

The types of conditions/transients that need to be simulated include normal operation, operational events, design basis events and beyond design basis events. The normal operation of the plant deals with start-up, power operation and load following. The operational events include transients such as reactivity insertion or undercooling transients, possible sodium-water interaction in the steam generating system, interruption in the off-site power supply or the feedwater supply system. The design basis events are essentially what the plants are designed to accommodate and the last category primarily includes the core disruptive events. All of these transients are part of the standard safety analysis.

This session emphasizes the simulation of LMFBRs for only two key categories of the above-mentioned transients: operational disturbances or events and the post-shutdown decay heat removal. There are two papers from France on Superphenix 2, one from West Germany on their experience with the SSC code for SNR-300, one from Argonne on EBR-II control simulation, one from Brookhaven on the thermohydraulic modeling and simulation, and the last one from Argonne on their effort to adapt the SAS (whole-core accident code) to simulate the entire plant. In addition, there is a paper on the engineering simulator from Japan, but it is to be given in another session.

In the following, the author has attempted to give an overview of the current state-of-the-art and future directions. These primarily reflect the results obtained within the United States.

State-of-the-Art

The subject of dynamic simulation of LMFBRs has recently been reviewed by Agrawal and Khatib-Rahbar (1). They have, in this paper, discussed thermohydraulic models that are being used in the fast breeder reactor simulation for operational transients as well as for the shutdown heat removal. The latter subject has also been the topic of an international specialist's meeting at the Brookhaven National Laboratory in 1980 (2).

Some of the earlier U.S. efforts in fast reactor simulation were dedicated specifically for a plant. For example, the IANUS code (3) was developed for the Fast Flux Test Facility, DEMO (4) for the Clinch River Breeder Reactor (CRBR) and NATDEMO (5) for the EBR-II. These codes provided overall characterization of their respective plant. Detailed hot-spot temperatures in core assemblies were then obtained by employing another computer program in tandem. This two-stage technique has worked acceptably, but it does suffer from some drawbacks. More pertinent drawbacks are 1) the plant response code is not affected by the outcome of detailed sub-component analysis, 2) need to perform calculations in tandem and 3) cannot handle situations where the coupling between the plant and detailed codes is not weak (such a situation may arise if there is coolant flow reversal or coolant boils in some of the assemblies). In addition, these codes generally do not account for flow-dependent fiction losses in the reactor core and the plant. Finally, these codes are hard-wired, hence cannot be used without significant modifications. These difficulties have been rectified in the SSC series of codes (6). The SSC-L code is being extensively used by a number of users both at home and overseas. It is also being used in connection with the CRBR licensing.

Some comments on the sodium boiling are in order. For most operational events, sodium boiling, even in the hottest channel, is precluded by a substantial margin. On the other hand, for some postulated events, even with the plant protection system operating per

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design, there exists a possibility for boiling in hot fuel or blanket channels. Typically, the reactor power is at decay heat levels (< 8% of full flows) corresponding to the natural circulation. This situation is characterized as low flux-low flow. The mechanism of boiling under these conditions is quite different from those encountered in the whole-core disruptive accidents (CDAs). Coolant boiling for the CDAs are characterized as high flux-high flow. Thus, the slug or multiple slug model for sodium boiling used, for example, in the SAS code (7) would not be appropriate for the shutdown conditions. A homogeneous boiling model, therefore, was incorporated in the SSC code (8).

Unresolved Issues

There seem to be two key unresolved issues: code verification and the accident management. The computer codes in use today are quite sophisticated and large. The simulation of physical processes is done by making a number of convenient approximations and assumptions. It is therefore mandatory to verify the computer codes to ascertain the modeling and programming adequacies. This can be accomplished by applying either the code or the models used in the code to pre- and post-prediction of the pertinent tests (whole plant as well as separate effect tests). At the same time, an acceptance criterion should also be defined. A close cousin of the code verification task is to identify sensitive and important parameters. Finally, the code verification is a continuing process - each application to a test results in a data point and as these data become large, one asymptotically obtains a verified code.

The accident management is a relatively newer item in the list of simulation. So far, most of the whole-plant simulation effort is geared for containing the consequences of an event. But, in view of the TMI-2 experience, some simulation for events which may not be "contained" within the plant boundaries must be undertaken. One is interested in simulating, for example, the effects of accidental release of radioactivity.

Future Trend

Some of the important directions that are likely to be pursued in the near future are highlighted here. These are:

- The need for a passive mode of decay heat removal from LMFBRs is likely to dominate the safety issues. Therefore, the predictability of the shutdown heat removal from an intact (nominal) as well as partially damaged plant should be assured with a high level of confidence.
- Many laboratory tests as well as in-plant tests may be conducted to verify (and possibly modify) computer codes.
- Some probabilistic risk assessment techniques may be utilized to estimate frequencies of occurrence of normal and off-normal events. Then, with the help of deterministic analysis for these events, the cumulative load on structure may be quantified.
- Finally, the international cooperation in exchanging data may get further support since experiments are expensive and time consuming.

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MEMO FOR: Jim McKnight, DMB

FROM: P. Larkins, TIDC
J. Resner, TIDC

SUBJECT: Transmittal of Speeches

Attached are two copies of a speech to be sent to the PDR and TERA. We have filed the NRC Form 426.