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The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: NRC RESEARCH RELATED TO HEAT TRANSFER AND FLUID TRANSPORT IN
NUCLEAR POWER PLANTS

During the 338th meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1988, we considered a report from our Subcommittee on Thermal Hydraulic Phenomena pertaining to its review of research activities sponsored by the NRC on reactor thermal-hydraulic phenomena. This subject had been considered during the 337th ACRS meeting, May 5-7, 1988 and a number of previous meetings of the ACRS and the Subcommittee. We also had the benefit of the documents referenced.

Background

The technical subjects of heat transfer and fluid transport are of cardinal importance when considering the safety of nuclear power plants. They are of chief concern in relation to LOCA and reactor transients, and the performance of ECCS, steam generators, secondary systems, and containment phenomena. These issues have been studied extensively in experimental and analytical programs sponsored by NRC and the industry.

The earliest reactor safety research dealt principally with reactivity addition accidents. Later research, sponsored initially by the Atomic Energy Commission and then by the NRC, was devoted almost exclusively to providing a better understanding of the hypothetical large-break LOCA (LBLOCA) and the related performance of ECCS and containment systems. It was soon perceived that the complexities of two-phase flow and the time sequences involved in a LBLOCA were such that straightforward experimental representation was difficult. It was further perceived that traditional methods for application of empirical data to plants were subject to challenge. In response to this, the NRC sponsored development of complex computer codes at the Department of Energy's (DOE's) national laboratories.

These codes were intended to provide consistent treatment of the relationships among various plant systems during rapid transients and to bridge gaps in data from the various test programs. In time, as the physical representations of two-phase flow and heat-transfer phenomena and the plant systems were made more detailed, broad interpolation and extrapolation from experiments were attempted and came to be relied upon.

This general strategy, that is, primary dependence on detailed mathematical models of physical phenomena coded for rapid analysis by

computers, has been adopted by the NRC for studying other technical areas involving complex phenomena and interactions.

Another product of this era, in addition to the extensive base of experimental information and the codes, has been a skilled cadre of experts. These experts can be found in the NRC, the national laboratories, the universities, and the industry. These people understand thermal-hydraulic phenomena associated with LBLOCA probably as extensively as almost any other similar subject in modern technology is understood. These experts are also well-schooled in the general strategy described above. It is important that a cohesive group of experts be maintained.

The next period of thermal-hydraulic research followed the accident at TMI-2 and the gradual assimilation of the perspectives provided by probabilistic risk assessments. Interest shifted toward small-break LOCA and plant transients. De-emphasis of LBLOCA began. The large system codes, developed in the previous era, were available and, with modification, came to be the primary means by which these less dramatic reactor events were analyzed. It was recognized that these codes, originally written to incorporate certain conservatisms in an attempt to envelop uncertainties in LBLOCA analyses, would serve their new purposes only if they could more realistically track the evolution of transients. The conversion to realistic or "best estimate" codes is now complete.

In each of these periods of thermal-hydraulic research, scientific interest was confined to phenomena and time sequences associated with normal plant conditions and with faulted conditions extending to, but not beyond, the point at which a coolable core geometry is lost; research activity also included consideration of single- and two-phase flow, heat transfer, and nonequilibrium conditions.

General Recommendations

The following comments include recommendations for future research in the traditional "thermal-hydraulic" area, including specific recommendations for the code development program.

For the sake of discussion we have posed our comments in this section as a series of questions, with our recommendations following as answers:

~ Is there a need to continue a program of experimental research in the traditional thermal-hydraulic area?

Yes, but not indefinitely nor without specific purpose. There are several matters that currently need attention and will require several more years of experimental work at a moderate rate. Specific recommendations will be given below.

NRC has played a key role in the fundamental development of thermal sciences related to nuclear power plants. It should continue to furnish leadership, perhaps by more clearly defining basic research needs or directions for the DOE and industry.

~ Is the strategy of dependence on large system codes as primary

tools for analysis valid?

This strategy has both strength and weakness. As strength, the system codes have the ability to model, in a consistent and reasonably accurate way, the dynamic relationships among the various elements in a plant heat transport system. They are weaker in the accuracy with which they model the complex physical behavior of system subelements, especially in extreme off-normal conditions. This weakness becomes an important problem because analysts and decisionmakers tend to overlook the inaccuracies and to behave as if the codes were revealing physically correct and validated information about the plants. These codes are also very expensive to use and require specialists to use them properly.

These codes can be useful if they are regarded as simply one input, albeit often an important one, to the understanding of plant transients. The codes can be dangerously misleading if they are used without engineering judgment and to the exclusion of simpler but less comprehensive analyses. We are concerned that those conducting research in severe accident phenomena have fallen into this trap.

~ Should traditional (i.e., LOCA) code development be continued?

The codes are now adequate for the purposes for which they are needed and further development is unjustified. First, they satisfy the regulatory need related to the ECCS rule. For this the Code Scaling, Applicability, and Uncertainty (CSAU) Program is helpful. Second, they are adequate as general-purpose tools for exploring and gaining understanding of other plant transients, from a safety rather than a regulatory perspective. In this use, analysts should be guided by the comments above.

In making this recommendation, we recognize that the codes are not without flaws. However, we believe that not all of the imperfections in the codes can possibly be corrected by any reasonable program of research and code development. Marginal improvements that could be made over the next few years by extrapolating the recent levels of development work will not be sufficient to attain a significantly higher plateau of code accuracy and validation. The code development effort has been a substantial technical achievement and the codes have made an important contribution to nuclear power plant safety. Further refinement is unnecessary.

The CSAU Program will provide a reasoned perspective on the accuracy of the existing codes. With that perspective available, we endorse the general strategy proposed by the RES staff toward maintenance of existing codes. This would provide for completion of RELAP-5 and TRAC-PWR development through the International Code Assessment Program Consortium in 1989. A modest level of resources would be provided to maintain the codes overall (including TRAC-BWR, COBRA-NC, and RAMONA-3B), based on regulatory needs.

Nuclear power plants are complex machines, even in normal operating modes; they have many interrelated systems and processes. We believe that computer codes can model normal operating behavior accurately and usefully, if extreme physical phenomena are not

involved and if the codes can be validated by comparing their results to measurements of plant operating parameters. There is a significant resource in code development expertise at the national laboratories. Consideration should be given to using this resource with an approach to code development that takes advantage of inherent strengths in the present codes. Efforts should be concentrated on including all of the plant systems, providing code versions validated for specific plants and providing modeling and interfacing that is transparent and understandable for use by those expert in plant operation rather than just those expert in analysis by computers.

~ Is it essential that a cadre of experienced people be maintained?

It is essential to maintain such a cadre, because questions of fluid and heat transport will always be central to reactor safety. The NRC should maintain a center of expertise in experimental and analytical research in thermal-hydraulic phenomena. The Technical Support Center at the Idaho National Engineering Laboratory serves this purpose. However, the NRC should limit the program to: (1) confirming selected information supplied by industry and (2) exploring important issues that the industry is not addressing. Involvement of universities and other nongovernment research organizations should be encouraged. There should be free exchange of information with industry and international experts.

Specific Recommendations

- (1) The CSAU method, or something similar, can be used in other areas of safety analysis, that is, beyond the currently conceived purpose of assessing uncertainty associated with calculations by thermal-hydraulic codes. In particular, its application to severe accident studies and risk assessments could serve, not only to provide an improved perspective on uncertainty, but also as a guide to allocation of research resources. This should be investigated.
- (2) The current programs of research on B&W reactor systems and once-through steam generators should be continued only to the point that the technical understanding of B&W systems is comparable to that of other nuclear steam supply systems. In particular, it should be demonstrated that adequate capability for predicting B&W system performance is in hand.
- (3) Analysis of industry experience with water hammer events suggests that water hammer is not a significant initiator of nuclear power plant accidents. However, insufficient consideration has been given to whether water hammer, occurring as a consequence of other initiators, might contribute to unexpected failures that could compromise core cooling. This issue should be investigated.
- (4) The recent steam generator tube rupture (SGTR) at North Anna has been explained as the result of a series of mechanisms which indicate that multiple SGTRs are no more probable than has been believed. The licensee's technically complex explanation was based on poorly understood phenomena. The NRC should explore this issue sufficiently to confirm the licensee's conclusions.

- (5) Although the feed-and-bleed cooling process is not directly required by regulations, it is given credit in assessing the overall safety of individual plants and of the population of plants in the United States. The contribution made by feed-and-bleed cooling to the safety of plants needs to be better established. It is regarded as a "last ditch" cooling mode that can be effective in some plants. Risk assessments are ambiguous about its importance. There is significant uncertainty about the reliability with which this process can actually be carried out in many plants, perhaps in most. In particular, there are questions about the flow capacity and reliability of the valves (usually power operated relief valves) essential to provide bleed flow and blowdown quenching capacity. In addition, the complex flow path and the effects of uncovering the core do not seem to be well understood for all plants. Research should be directed toward resolving the key uncertainties related to providing assured feed and bleed at plants that depend on the process for a margin of improved safety.
- (6) The LBLOCA, the design-basis accident for certain plant systems, should be reconsidered in view of the results of research on leak-before-break and the revision to General Design Criterion 4. Thermal-Hydraulic research will be necessary in support of this effort.
- (7) The designs for so-called evolutionary LWRs and especially the "passive" LWR being developed by the Electric Power Research Institute and DOE, will require research by the NRC to confirm certain favorable characteristics being claimed. The DOE Advanced Reactor Severe Accident Program is not sufficient for this purpose. The NRC should use existing codes to review these designs so there is sufficient lead time to conduct more experimental or code development work, if necessary.
- (8) There is some uncertainty about applicability of the RELAP-5 code to BWRs and to LBLOCAs. This should be resolved.
- (9) Full documentation should be completed for the NRC codes that are maintained for active use. This should include not only user manuals but developmental assessment reports and "models and correlations" documents. Ideally, these would be published as NRC documents in the NUREG series to ensure widespread availability.
- (10) Thorough analyses have generally been made only for the initial period of reactor accidents such as LOCAs. Analyses of the follow-on transition to stable long-term cooling have been less comprehensive. We recommend that NRC determine whether a more systematic and complete study of the reliability of such transitions should be undertaken.

ACRS Members William Kerr, Harold Lewis, and Forrest Remick did not participate in the review of this matter.

Sincerely,

David A. Ward
Acting Chairman

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