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(Friday, March 22, 1996)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received two attached reports from its Advisory Committee on Reactor Safeguards (ACRS). The reports, in the form of letters, provide comments on:

--Recent probabilistic risk assessments performed by Brookhaven National Laboratory on fires and certain fire barrier issues; and

--Use of individual plant examinations in the regulatory process.

In addition, the NRC's executive director for operations received two ACRS reports. They provide comments on:

--An NRC program assessing the adequacy of a computer code for simulating the behavior of the Westinghouse Electric AP600 advanced pressurized water reactor design; and

--Resolution of generic safety issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System."

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Attachments:
As stated

March 15, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: REVIEW OF RECENT FIRE PROBABILISTIC RISK ASSESSMENT REPORTS BY BROOKHAVEN NATIONAL LABORATORY AND CERTAIN FIRE BARRIER ISSUES

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed scoping fire probabilistic risk assessments (PRAs) performed by Brookhaven National Laboratory (BNL). We had the benefit of discussions with representatives of the staff, BNL, and the National Institute of Standards and Technology (NIST). Our Subcommittee on Fire Protection discussed this matter during a meeting on February 29, 1996. We also had the benefit of the documents referenced.

At your request, we reviewed both the PRA model that evaluated the strategy of using self-induced station blackout (SISBO) to mitigate the consequences of a fire in the control room or cable spreading room and the PRA-based scoping analysis of degraded fire barriers. We also discussed the development of alternate time-temperature curves for qualification of fire barriers and the status of other fire protection issues.

To comply with Appendix R requirements, eight units have procedures that require initiating a station blackout (SBO) condition. An additional fifteen units have procedures for dealing with fires in critical areas that could result in an SBO. The PRA by BNL evaluated the effects of different schemes for managing the electrical systems in the plant when a fire in the control room has required use of the alternate shutdown panel.

The study focused on the effectiveness of the procedures used to mitigate the fire and did not address the probabilistic treatment of fires. The scope of the study did not include a number of issues that could affect the conclusions. For example, the BNL study addressed neither the effects of fire and smoke on human actions nor the possible damage to sensitive electronic control and safety instrumentation. The study is weak in the areas of modeling human actions for the manual shutdown and restart of electrical equipment after an SBO condition. Because of the limitations of the analysis and the failure to quantify uncertainties, no substantive conclusions can be drawn from this scoping study. The limitations of the analysis should be addressed in Phase 2 of this study. A meaningful uncertainty analysis should also be performed.

In the analysis of degraded fire barriers, BNL developed core-damage frequencies for fire scenarios involving failures of fire protection features such as cable tray fire barriers, automatic detection and suppression systems, and fire barrier penetrations. The PRA model did not examine degrees of fire barrier degradation.

The analysis was based on event tree/fault tree models. Although this is a step in the right direction, the analysis does not use the best available methods for modeling fire propagation, detection, and suppression. It does not model the fundamental competition between the time to damage and the time to detection/suppression. Most current fire PRAs have adopted the competing processes model.

We also discussed the program proposed to the staff by NIST to develop alternate time-temperature curves for nuclear power plant fire barrier qualification. The program includes development of models, ASTM E119-type full-scale furnace tests, and test methods to simulate barrier response. We question the need for this program. We have been told that alternate time-temperature curves have been produced by the insurance industry. Furthermore, a large number of fire models exist, some of which are being evaluated by the Department of Energy. Although the need for new models is not clear, more validation of these models with experimental data is needed. Some data exist (NUREG/CR-6017). Comparisons with fire model simulations show that the results are very sensitive to input parameters that are not always well known.

The staff summarized the progress of licensee actions to correct deficiencies associated with Thermo-Lag fire barriers. The program appears to be meeting its objectives.

Sincerely,

/s/

T. S. Kress
Chairman

<u>References</u>:

1. Brookhaven National Laboratory, Draft Technical Letter Report, FIN L-2629, "Risk Evaluation of the Response of PWRs to Severe Fires in Critical Locations," May 30, 1995 (Draft Predecisional)
2. Brookhaven National Laboratory, Technical Evaluation Report, FIN L-1311, "A Risk-Based Approach for Evaluation of Fire

Mitigation Features in Nuclear Power Plants," November 21, 1995 (Draft Predecisional)

3. U. S. Nuclear Regulatory Commission, NUREG/CR-6017 and SAND93-0528, "Fire Modeling of the Heiss Dampf Reaktor Containment," September 1995

March 8, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: USE OF INDIVIDUAL PLANT EXAMINATIONS IN THE REGULATORY PROCESS

During the 428th and 429th meetings of the Advisory Committee on Reactor Safeguards, February 8-10 and March 7-9, 1996, respectively, we discussed the Individual Plant Examination (IPE) review process and findings with the NRC staff. Our Subcommittee on IPEs also met with the staff and its contractors on January 26, 1996, to review this matter. We also had the benefit of the documents referenced. This report is in response to the December 27, 1995 Staff Requirements Memorandum (SRM).

In the SRM, the Commission requested "the ACRS views on the extent to which the current spectrum of IPEs can be used in the regulatory process." We interpret this request as referring to potential regulatory uses of the IPEs that were not delineated in Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." This report includes comments on both the Generic Letter goals and the Commission request.

Goals of Generic Letter 88-20

The purpose of the IPE program, as stated in Generic Letter 88-20, was for each licensee:

(1) to develop an appreciation of severe accident behavior

(2) to understand the most likely severe accident sequences that could occur at its plant

(3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases

(4) to reduce, if necessary, the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and proce-

dures that would help prevent or mitigate severe accidents.

We note that the IPEs were to be limited to the examination of internal initiating events and internal floods with the reactor at power and that individual and societal risks were not to be estimated. Other programs deal with external events and shutdown risk.

The IPE program has been successful at most utilities in meeting goal (1) and, to a lesser extent, goals (2) and (3) of the Generic Letter. Goal (4) of the Generic Letter also appears to have been achieved. We were told that most licensees discovered weaknesses and took corrective actions. In addition, this program has been beneficial in educating a broader segment of the NRC staff about the issues related to these goals.

We were told by the staff that all licensees submitted a Level-1 probabilistic risk assessment (PRA). Most licensees also submitted a Level-2 PRA, although some addressed Level-2 phenomena in a rudimentary manner. The methods and data sources used by different licensees varied widely. In some cases, the choices appeared to be arbitrary. Some licensees chose to include common-cause failures only for major components, while others chose to ignore them completely.

It is difficult to determine the extent to which the variability in IPE results for similar classes of plants is due to actual plant differences or to modeling assumptions. Although some of the causes for this variability may be immediately apparent, others are not. The latter include assumptions made about success criteria, the assumed dependencies between operator actions, and the level of decomposition in fault-tree analyses. (We note that the fault trees were not requested as part of the IPE submittals.)

An example of a potentially significant impact of modeling differences is the range of core-damage frequencies (CDFs) for BWR 3/4s that the staff has compiled. This range is from about 10^{-7} to about 10^{-4} per reactor-year. Although the staff has stated that such differences are primarily due to plant differences, this range of results seems unrealistic given the similarity among BWR 3/4s.

Use of IPEs in the Regulatory Process

As discussed above, the quality and consistency of the IPEs vary and the impact of assumptions and analytical models is difficult to assess. On a case-by-case basis, however, additional and extended use of these IPEs is possible. As specific regulatory issues arise, the PRA Standard Review Plan now being developed by the staff can serve as a template for judging the quality and acceptability of the individual plant PRA for the proposed application.

As the agency moves toward risk-informed regulation, there will be an increasing need for full-scope PRAs that incorporate fire risk, external events, other modes of operation, and site-specific consequences. When requests for risk-informed regulatory action arise, the NRC staff should make it clear that a relevant PRA should be used.

To achieve these goals, especially consistency, some degree of standardization will be required. Standardizing PRA models and methods has been a controversial subject. Proponents argue that it would create a basis for comparison of PRA results, while opponents fear that it would inhibit methodological developments. We recommend that IPEs be reviewed to identify acceptable and unacceptable assumptions and/or models. Codification of assumptions and models ought not inhibit the continued development of PRA methods. These activities would be a significant first step toward addressing the Commission's statement in the SRM dated June 16, 1995, "that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis."

We believe that the NRC could make additional use of the present IPEs (except those that the staff has found to use unacceptable methods or models) for a limited number of applications (e.g., regulatory analyses and prioritization of generic issues).

The staff stated that the CDFs for several PWRs are greater than 10^{-4} per reactor-year. Several BWRs have CDFs that are very close to 10^{-4} per reactor-year and the conditional containment failure probabilities for BWR Mark I containments range from about 0.02 to about 0.6. Although the PRAs have limitations as discussed above, these numbers suggest that an investigation would be warranted to reassess their validity and to verify that the very low numbers reported by some other plants reflect actual plant differences.

Our conclusion is that the IPE program has met successfully the objectives of Generic Letter 88-20. This program has developed a risk awareness, both in the utilities and the NRC, that will contribute significantly to efforts to establish a risk-informed and performance-oriented regulatory system. The plant-specific

IPEs are an extremely valuable asset that should not be permitted to languish unimproved and unused.

Sincerely,

/s/

T. S. Kress
Chairman

References:
1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary, NRC, to the File regarding Meeting with ACRS on June 8, 1995
2. Staff Requirements Memorandum dated December 27, 1995, from John C. Hoyle, Secretary, NRC, to John T. Larkins, ACRS regarding Meeting with ACRS on December 8, 1995
3. Generic Letter 88-20, dated November 23, 1988, to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, Subject: Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)

March 19, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor

SUBJECT: NRC STAFF PROGRAM ON THE ADEQUACY ASSESSMENT OF THE RELAP5/MOD3 CODE FOR SIMULATION OF AP600 PASSIVE PLANT BEHAVIOR

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed the program being conducted by the Office of Nuclear Regulatory Research (RES) to assess the adequacy of the RELAP5/MOD3 code for simulating the behavior of the Westinghouse AP600 passive plant design. During this review, we had the benefit of discussions with representatives and consultants of the NRC staff and the Idaho National Engineering Laboratory (INEL). Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on this matter on February 22-23, 1996. We also had the benefit of the referenced documents.

We have been asked to comment on the approach and methodology for demonstrating the adequacy of the RELAP5/MOD3 code to calculate AP600 passive plant behavior in support of the design certification review. We believe that the overall approach and methodology being employed by RES for this assessment is acceptable. Most of the necessary elements are in place. A substantial amount of work remains, however, and we believe that the schedule for successful completion cannot be met.

Our comments and recommendations relative to this review, primarily based on oral presentations, are:

- Since we last reviewed this program in 1994, significant improvements have been made. The most significant has been the increased emphasis on the code improvement program. Other changes that have led to excellent results include the involvement of outside technical expertise, via the Thermal Hydraulic Expert Consultants group and the direct involvement of RES technical personnel in the research activities. Particularly noteworthy accomplishments include the analysis of water hammer, the treatment of flow oscillations observed in the tests during injection from the In-containment

Refueling-Water Storage Tank and the evaluation and explanation of strong thermal stratification in the ROSA cold leg.

- RES should perform a more robust and complete top-down system scaling analysis for ROSA, SPES, and OSU. An entire transient should be evaluated to quantify the effects of various distortions in the three facilities and to demonstrate that the experimental database is sufficient to validate the code. Any additional distortions or anomalies identified should be added to the list of distortions compiled by RES in late-1994, and that remain to be addressed. The scaling effort should be integrated with the Phenomena Identification and Ranking Table.

- The thermal stratification that was seen in ROSA tests for a one-inch cold-leg break was initially identified as a potentially important safety issue for the AP600. It has now been shown to be just a manifestation of scale distortion in the ROSA facility. This demonstrates the need to identify and explain anomalous behavior.

- The thermal stratification in the Core Makeup Tank (CMT) observed in the tests needs to be studied. Its effects on core inventory have to be understood because neither RELAP5/MOD3 nor the Westinghouse computer codes can, at present, reliably predict thermal stratification.

- The screening study for water hammer in the AP600 design addressed an important safety issue. The study allows an analysis of the potential for such events and provides a method for estimating the resulting loads in susceptible areas. We recommend that this study be published soon as a separate report.

- The documentation provided for our review did not, by itself, furnish an adequate basis upon which we could logically endorse the process. The documentation provided to the Thermal Hydraulic Phenomena Subcommittee in advance of the February 22-23, 1996 meeting was inconsistent and contained results declared incorrect by RES during the meeting. Furthermore, the RELAP5/MOD3 Code Manual published in August 1995 was not provided to us in time to support our review.

- RELAP5 is still undergoing significant and rapid modifications. A calculation has not yet been performed with a version of the code that contains all the planned changes. Numerous calculations will need to be performed to mature the code and validate it using data obtained from various separate effects and integral facilities tests.

Overall, the approach and methodology for qualifying RELAP5/MOD3 for AP600 simulation appear to be adequate. However, two possible "show stoppers" remain: 1) simulation of the CMT thermal stratification and 2) simulation of long-term cooling, which is still an issue. Serious consideration should be given to addressing these obstacles.

Dr. George Apostolakis did not participate in the Committee's deliberations of this matter.

Sincerely,

/s/

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated January 22, 1996 from M. W. Hodges, Office of Nuclear Regulatory Research, NRC, to J. Larkins, Advisory Committee on Reactor Safeguards, NRC, transmitting:

    - Volume 2 of 10 volumes of adequacy demonstration reports, "Adequacy Assessment Overview"

    - Idaho National Engineering Laboratory draft report prepared for U.S. Nuclear Regulatory Commission, "Adequacy Evaluation of RELAP5/MOD3 for Simulating AP600 Small Break Loss-of-Coolant Accidents, Volume 2: Horizontal Integrated Analysis of the AP600 1-Inch Diameter Cold Leg Break," November 1995, with Appendices A-K (Proprietary)

2. Idaho National Engineering Laboratory, draft report prepared for U.S. Nuclear Regulatory Commission, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," January 1996

3. Letter report dated April 12, 1995, to James M. Taylor, Executive Director for Operations, NRC, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Review

4. Letter dated May 8, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: Staff Response to ACRS Letter Dated April 12, 1995, on NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Reviews

March 14, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE 78, "MONITORING OF FATIGUE TRANSIENT LIMITS FOR THE REACTOR COOLANT SYSTEM"

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we completed our deliberations on the resolution of the subject Generic Safety Issue that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

This Generic Safety Issue was originally developed to determine whether licensees need to perform transient monitoring to ensure compliance with requirements concerning fatigue failure. The transient monitoring concern was subsumed in the Fatigue Action Plan, which was reported as complete in SECY-95-245, "Completion of the Fatigue Action Plan."

The current scope of the Generic Safety Issue is focused on the evaluation of risk from fatigue failure. The staff completed a study that demonstrated that the risk from fatigue failure of the primary coolant pressure boundary components is very small. The analyses used in the study were based on the assumption that the probability of crack initiation by fatigue in a component subject to cyclic loads and the probability of crack propagation through the wall are independent. The product of these probabilities was used to calculate the change in core-damage frequency caused by fatigue failure of a component.

The analyses, as presented to us by the staff to demonstrate its conclusion, lacked sufficient detail to be convincing. Additional discussions with the staff demonstrated that more complete analyses using the PRAISE code have led to the same conclusion. The PRAISE analyses of the failure probability of primary system piping assumed that a distribution of cracks existed in a component and calculated the probabilities of crack propagation through the wall and failure. Parametric studies using the PRAISE code showed that the calculated probabilities of failure are small, even when very conservative loads and flaw-size distributions are assumed. The staff provided a careful quantification of uncertainty of fatigue crack initiation. We recommend such consideration of uncertainties in any future analyses regardless of the technical approach adopted.

We believe that the staff's conclusion concerning the risk significance of fatigue failure of reactor components is correct. Thus, we agree that this Generic Safety Issue is resolved.

Dr. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/s/

T. S. Kress
Chairman

References:
1. Memorandum dated August 18, 1995, from Charles Serpan, Jr., NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, Subject: Proposed Resolution of Generic Safety Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System"
2. SECY-95-245 dated September 25, 1995, from James M. Taylor, Executive Director for Operations, to the Commissioners, Subject: Completion of the Fatigue Action Plan
3. Memorandum dated October 27, 1995, from Jeff Keisler and Omesh Chopra, Argonne National Laboratory, to Craig Hrabal, NRC Office of Nuclear Regulatory Research, Subject: Uncertainty Estimates for the Probability of Fatigue Crack Initiation in Reactor Components, NUREG/CR-6335, ANL-95/15
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6237, "Statistical Analysis of Fatigue Strain-Life Data for Carbon and Low-Alloy Steels," August 1994
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," June 1995