CORRECTED COPY: 2/8/96
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 15, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED COMMISSION PAPER ON STAFF POSITIONS ON TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we discussed the subject Commission paper. Our Subcommittee on Westinghouse Standard Plant Designs met on May 31, 1995, to review this matter. During these meetings, we had the benefit of discussions with representatives of the staff and Westinghouse. We also had the benefit of the documents referenced.

The intent of the proposed Commission paper is to record the staff positions on ten separate issues. In some cases, however, the reviews have not progressed to the point that the staff can recommend a position. In such cases, the paper describes the approach that Westinghouse is proposing in its application with little staff comment. The staff is continuing its review of these matters.

Our comments follow the same organization found in the attachment to the paper.

I. Leak-Before-Break Approach

Westinghouse proposes that any dynamic effects associated with postulated pipe ruptures in a broad range of pipe sizes can safely be excluded from the AP600 piping design basis by virtue of the current understanding of leakage and flaw sizes, and the proposed leakage rate limit of 0.5 gpm. The range of pipe sizes (4 inch diameter and greater) that would be covered by the leak-before-break (LBB) approach is broader than that allowed in currently operating pressurized water reactors for which the usual plant leakage rate limit is set at 1.0 gpm.

The staff agreed that the leakage rate limit of 0.5 gpm is achievable in the AP600 design but wishes to add conservatism in applying the LBB approach at the design certification stage by requiring that all loads used in the piping design be multiplied by a factor of 1.4. The staff considers this prudent because the detailed design of piping configuration

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and the as-built stress levels will not be available for review at the certification stage. Westinghouse argued that this added conservatism is not needed and will act to limit the gains in plant arrangement, economy, and safety that application of the LBB approach could provide.

We believe that the staff is hard pressed to justify adding conservatism on all the piping loads above that which has been applied to other plants. Although it is true that the details of the piping design are some years away, the staff and Westinghouse should now be able to combine the standard piping design protocols with what is known about the performance of flawed pipes into a design criterion without excessive conservatism.

II. Security Design

The proposed AP600 plant arrangement includes a vehicle barrier at a "stand-off distance," but the personnel access control will be located within the nuclear island of the plant. The vital areas of the plant are coterminous. This feature is not specific to the passive nature of the plant design and might be offered in other plant designs as well. The staff continues to review the proposed design, but seems receptive to the idea. The staff believes that inspections, tests, analyses, and acceptance criteria (ITAAC) may be required for this security design.

We believe the proposed security design could meet the safety and security requirements when implemented, and we are interested in the continuing staff review of the proposed design. We also noted that the design seems to offer less flexibility for the many work access points that operating plants need during outage periods.

III. Technical Specifications

Westinghouse proposes that hot shutdown, rather than cold shutdown, be considered the safe shutdown end state. The staff evaluation has not progressed to the point where the staff could make substantial comment. We also will withhold comment at this time. We expect that review of the probabilistic risk assessment regarding this issue will be instructive.

IV. Initial Test Program

Westinghouse and the staff have been discussing the content of the initial test program to be performed by the first plant built under the design certification, and test programs to be performed by subsequent plants. We believe that the staff is approaching the matter appropriately. When the discussions have resulted in new submittals from Westinghouse, we may have more information on which to comment.

V. Passive System Thermal-Hydraulic Performance Reliability

The staff believes that the magnitude of the natural forces relied on for the passive safety systems leads to large uncertainties in the thermal-hydraulic performance. It stated that one could quantify these

uncertainties, but only with "a prohibitively large number of computations." The staff proposed instead that a surrogate conservative risk-based margins approach be developed to eliminate the need to quantify thermal-hydraulic uncertainty for most, if not all, accident sequences.

This approach may be expedient, but we believe efforts should continue on the quantification of the uncertainty for use in probabilistic risk assessments.

VI. Regulatory Treatment of Non-Safety Systems

Westinghouse and the staff have been meeting to review the need for some level of regulatory treatment for systems and components that are not safety grade, but that have important support and backup functions. A key issue identified by the staff in this regard is the reliance that Westinghouse places on equipment or materials that may be required beyond 72 hours following an accident but which are not to be stored onsite. The staff review of this issue is currently under way, and the staff has not stated a position beyond identifying concerns.

Accident scenarios for existing plants reach a point when reliance must be placed on offsite materials. We expect that the staff will need to be satisfied that the AP600 design can be brought to a stable condition using onsite equipment, and that any additional needed resources are reasonably available.

VII. Containment Performance

The staff intends to use both deterministic and probabilistic containment performance goals in reviewing the AP600. This is consistent with the Commission direction given in the July 21, 1993 Staff Requirements Memorandum related to SECY-93-087. We believe that the staff position is appropriate.

VIII. External Reactor Vessel Cooling

Westinghouse proposes a severe accident mitigation strategy for the AP600 that includes the ability to flood the cavity under the reactor to a level that is effective in cooling the lower reactor vessel shell and preventing reactor vessel melt-through following core melt. The staff stated that this would be a desirable feature if the technical issues can be resolved. The staff is pursuing those issues with Westinghouse. We believe that the staff is following an appropriate path, but we will closely follow the resolution of the technical issues.

IX. Passive Hydrogen Control Measures

The proposed AP600 design includes unpowered catalytic recombiners to control hydrogen generated in a design-basis accident (DBA). This is consistent with the overall concept of controlling design-basis accidents with passive measures. (The plan is to use igniters to control severe

accident hydrogen.) There are technical questions involving the qualification and effectiveness of catalytic recombiners in an accident environment. The staff proposes to approve the use of passive recombiners contingent on the resolution of these issues. We believe that the staff position is appropriate.

X. DBA and Long-Term Severe Accident Radiological Consequences

While the passive nature of the AP600 safety features is very attractive, the design has some downside characteristics. Post-accident pressure in the containment will remain positive longer than a plant designed with active cooling. Further, following severe accidents, the removal of radioactive species from the containment atmosphere is expected to be less efficient with passive means than it would be using active sprays or filters. Thus, there is the potential for radioactive leakage for an extended period, compared to that of the existing plants. The staff believes that this situation calls for consideration of additional means, such as a nonsafety-grade containment spray, to reduce containment pressure and suspended radionuclides following a severe accident. The staff has asked Westinghouse to reconsider its proposed position in this regard.

In addition, Westinghouse proposes a source term somewhat different from what the staff would use with respect to both timing and release fractions. The staff indicates that the technical differences here would not be of much concern if the staff can be satisfied that there would be an active system available to reduce the containment leakage potential.

We believe that the issues associated with the potential for radioactive leakage and the source term should be treated separately. We believe that the staff position on the source term is appropriate. The radioactive leakage from the proposed containment design, however, should be considered with respect to public risk and the safety goals.

In the course of this review, it has occurred to us that the certification of advanced light-water reactors provides an important opportunity to continue the evolution toward performance-based regulation. Current plans, unfortunately, do not take complete advantage of this opportunity, perhaps because of schedule constraints. The debate over the procedure to impose unquantified levels of conservatism on analyses of leak-before-break for small-diameter piping reflects a continuation of past practice. The aspirations of both the industry and the NRC would be better served by a performance-based criterion. Similarly, arguments on the time frame for analyses of radionuclide concentrations in containment would be unnecessary if a performance-based criterion were derived. In general, such performance-based criteria would be more consistent with the state-of-the-art engineering being employed in the design of advanced light-water reactors than the continued use of traditional criteria developed in the past when there was a poorer understanding of safety-related processes and phenomena.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding the severe accident source term. Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding external reactor vessel cooling.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated May 15, 1995, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Advance Information Copy of Forthcoming Commission Paper - Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
2. SECY-93-087 dated April 2, 1993, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs
3. SRM dated July 21, 1993, from S. Chilk, Secretary of the Commission, to J. Taylor, NRC Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs

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

June 16, 1995

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL POLICY STATEMENT ON THE USE OF PROBABILISTIC RISK ASSESSMENT METHODS IN NUCLEAR REGULATORY ACTIVITIES

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we reviewed the proposed final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities. We had the benefit of presentations by the NRC staff concerning the resolution of public comments as well as comments we made on a draft version of the Policy Statement. We also had the benefit of presentations by representatives of the Nuclear Energy Institute concerning a draft PSA Applications Guide. Finally, we had the benefit of the referenced documents.

We support a policy statement that encourages the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities. A policy statement that extends the use of such methods beyond the regulation of nuclear power reactors into other areas within the jurisdiction of the NRC provides a welcome opportunity to improve both the efficiency and the effectiveness of the body of the NRC regulations. Revisions made to the Policy Statement accommodate comments we made on an earlier draft. We feel it useful to issue a policy statement to update positions adopted in the past by the NRC concerning the use of PRA.

We are interested in the challenges that will have to be met to implement the Policy Statement. Technically defensible, risk-based regulatory activities will require the availability of PRAs that are adequately complete and of acceptable quality. Uncertainties in the results of these risk assessments will have to be characterized adequately. The staff indicated that it is aware of these needs. We look forward to hearing more about staff efforts to define standards for PRAs and strategies that will be adopted to audit and to review PRAs submitted by licensees.

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The staff is now considering the decision criteria that will be used in conjunction with the application of PRAs. The staff has stated that it feels inhibited from using the NRC safety goals in decisions concerning specific plants. We encourage the use of technically defensible PRA methods for risk management of individual plants consistent with the NRC safety goals. We note that, in such applications, these goals should not be treated as safety criteria. We believe that plant-specific risk management is an important subject which we plan to pursue. We will report on our findings in the future.

The widespread use of PRA methods within the NRC will necessitate a cultural change within the agency. The staff will have to be receptive to different approaches to given issues by different licensees. Training for the staff may need to be on more than PRA applications and methods. For instance, training in formal decision analysis methods may also assist the needed change in culture at the NRC. We are interested in the full scope of the training program in PRA being developed for the NRC staff. We plan to review this training program and the PRA research program that NRC supports.

The Policy Statement calls for the consideration of the use of PRA methods in areas where these methods have not heretofore been extensively used. Consequently, the methods for these new applications are not as well developed as they are for application to nuclear power plants. The NRC may need to support an expanded research effort in the development of PRA methods for application in these new areas.

Sincerely,

T. S. Kress
Chairman

References:

1. SECY-95-126 dated May 18, 1995, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (Draft Predecisional)
2. ACRS Report dated May 11, 1994, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities
3. Letter dated January 17, 1995, from William H. Rasin, Nuclear Energy Institute, to Ashok C. Thadani, Office of Nuclear

Reactor Regulation, NRC, transmitting final draft of PSA Applications Guide

4. ACRS Report dated May 13, 1987, from William Kerr, Chairman, ACRS, to The Honorable Lando W. Zech, Chairman, NRC, Subject: ACRS Comments On An Implementation Plan For The Safety Goal Policy

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

July 20, 1995

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

Dear Mr. Taylor:

SUBJECT: HEALTH EFFECTS VALUATION

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, we discussed the recent staff reconsideration of the health effects valuation. During this meeting, we had the benefit of discussions with representatives of the staff. We also had the benefit of the document referenced but it differs in some details from the presentation.

In reviewing the health effects valuation, the staff recognized the recent risk coefficients issued by the International Commission on Radiological Protection and retained the linear dose hypothesis. These were used along with the Office of Management and Budget (OMB) recommended value for a "statistical life" to arrive at an indicated increase from the present $1000/person-rem to $2000/person-rem. We were told that such a change is unwarranted because of the order-of-magnitude uncertainty in the regulatory analysis. Consequently, the staff is not proposing to change the value and is considering the following four options for proceeding on this issue:

- Retain the $1000/person-rem but require discounting.
- Retain the $1000/person-rem but require separate quantification of offsite property effects.
- Retain the $1000/person-rem but require both discounting and separate quantification of offsite property consequences.
- Retain status quo in the near term but allow use of the $2000/person-rem subject to discounting and/or separate quantification of offsite property consequences as part of optional sensitivity studies.

We believe that the change in the value is warranted and do not support any of the four options. In the interest of technical correctness, consistency in use across Federal agencies, and regulatory coherence, we recommend use of the new value of $2000/person-rem, as derived from the rounded-off product of the

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value of a "statistical life" ($3M) and a risk coefficient for the stochastic health effects (7.3 X 10^{-4} fatalities/person-rem). This value should be used as a dollar proxy for only the health effects associated with dose and should not be used (as in the past with the previous value) as a surrogate for other consequences such as prompt fatalities and land contamination. These other consequences should be evaluated separately as suggested in the draft Federal Register Notice. The MACCS code with an updated economic model would be an appropriate tool for such an evaluation. The new value should be expressed in terms of an identified year's dollars to allow users to make their own correction for inflation. Future effects should be discounted by present worth methods.

The selection of the value of a "statistical life" is the crucial determinant of the value of the health effects conversion factor. We believe that the present most appropriate means of establishing such a value is through the willingness-to-pay approach. This, however, can give a broad range of results that leads to a basic problem of defending the selection of any value from the range. The fact that a value is a median or a mean is not an appropriate defense for its selection in this case. In the absence of knowledge of any rationale underlying the existence of such a broad range, one has little recourse but to fall back on experience and judgment. In this spirit, we propose that there are basically two sound reasons for selecting the value of $3M for a "statistical life".

1. It is specifically cited by the OMB. This is a strong step toward consistency in use across government agencies.

2. Judgment and experience show that it is an appropriate value.

In the past, the $1000/person-rem has been used to represent both exposure and land contamination costs. We believe an exercise should be conducted to develop a sample estimate using the updated MACCS code for the relative magnitude of land contamination costs for severe accidents. Such a comparison would provide guidance on the need for a review of those previous decisions that may have involved predictions of considerable land contamination.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Reference:
Letter dated March 6, 1995, from Bill M. Morris, Office of Nuclear Regulatory Research, to T.S. Kress, Chairman, ACRS, Transmitting draft Federal Register Notice on Proposed Revision to the Health Effects Valuation. (DRAFT PREDECISIONAL)

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

September 15, 1995

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

Dear Mr. Taylor:

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we completed our discussion regarding the subject rulemaking petition. Our Auxiliary and Secondary Systems Subcommittee met on June 7, 1995, to begin the review of this matter. During these meetings, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI). We also had the benefit of the documents referenced.

The NEI petition for rulemaking proposes to amend 10 CFR 50.48, "Fire Protection," by adding an Appendix S, which is described as a "performance-based" alternative to the existing prescriptive Appendix R. NEI believes that the recommended addition to 10 CFR 50.48 will be "safety neutral" and that considerable cost savings will result.

We support risk-based regulations. It is not clear, however, how performance-based regulations should be developed from risk consideration. It is our perception that such regulations should include the following elements:

- Clearly stated objectives with demonstrable performance requirements, expressed either in deterministic or probabilistic terms.

- Flexibility in the methods that the licensee is permitted to use to meet the performance goals or criteria. These methods should be supported by operational experience and experimental results.

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- The regulatory body must have a valid means to establish that the performance criteria have been met.

Unfortunately, the proposed rule in the NEI petition is deficient in all these elements.

The objective of the proposed rule is to assure "that the safety functions required to safely shut a plant down and maintain it in a safe condition are maintained during and following a fire." It is further stated that fire modeling, as well as PRAs, may be used to identify the pertinent performance criteria. The proposed rule, however, avoids setting probabilistic requirements and uses non-quantitative language. Thus, there are references to "credible" fires and "credible" scenarios, as well as to "adequate" time for completing safety functions. These concepts need to be defined in quantitative, probabilistic terms. For example, we would expect a quantitative performance requirement for the probability that fire will compromise safe shutdown equipment and lead to core damage.

Some of the issues that the proposed rule raises could be naturally resolved in a PRA context. Examples are the inadvertent actuation of automatic suppression systems and the relevance of the current requirements regarding the concurrent occurrence of a fire and loss of offsite power. In addition, the proposed rule does not address the issue of transient fuels. PRAs have shown that, in some cases, transient fuels are required to produce fires of severity sufficient to damage redundant safety systems. Such transient fuels have been found in controlled areas in the past. Not only are transient fuels not addressed, the proposed rule suggests that some administrative controls dictated by Appendix R may be eliminated. We would prefer to see an evaluation of such issues in the context of a fire PRA.

We are concerned that neither the NRC nor NEI has any plans for conducting fire tests for refining the probabilistic analysis of time-to-suppression. We also have concerns about weakening the requirement for automatic fire detection systems, the lack of a methodology for treating the potentially damaging effects of smoke, the use of a limited fire initiation database, and the neglect of consideration of fire during shutdown. We will address these concerns should the rulemaking process advance.

Even though we support the use of PRA in the development of a performance-based rule, we note that, given the uncertainties in the state of the art, fire PRAs cannot be the sole basis for regulatory requirements. Developing the right mix of criteria based on PRA and criteria based on good engineering practice is a challenge and a necessary requirement for a well-written rule.

We believe it will take some time and resources to develop and institute performance-based fire regulation. We also believe doing so is an important step in the agency's move in this direction.

Additional comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton are presented below.

Sincerely,

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton

We support the Committee letter but have further comments for your consideration. The use of performance-based rules for fire protection is frustrated by conventional attitudes. The desire of regulators to have simple rules and tests for administrative convenience contrasts with the need of plant operators to have flexibility to arrive at optimal solutions. Unfortunately, the prescriptive characteristics embodied in regulations are accepted without proof, while any engineering solution supporting a performance requirement is subjected to a disproportionately higher standard of proof.

References:

1. Letter dated February 2, 1995, from W. Rasin, Nuclear Energy Institute, to John C. Hoyle, Acting Secretary, NRC, Subject: Petition for Rulemaking to Amend 10 CFR 50.48
2. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
3. SECY-95-034 dated February 13, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program
4. Memorandum dated December 30, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Eighth Quarterly Report on the Status of the Thermo-Lag Action Plan

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

September 15, 1995

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

Dear Chairman Jackson:

SUBJECT: DEVELOPMENT OF IMPROVED NONDESTRUCTIVE EXAMINATION (NDE) TECHNIQUES

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we heard presentations from representatives of the Electric Power Research Institute (EPRI), the EPRI Technical Advisory Group on NDE, Zetec, Babcock & Wilcox Nuclear Technologies, ABB-Combustion Engineering, and Westinghouse Electric Corporation regarding activities to improve NDE techniques for more accurately detecting and assessing steam generator tube defects. The status of staff activities on the development of a new steam generator rule and a supporting research program was also discussed. We had the benefit of the documents referenced.

In the June 16, 1995 Staff Requirements Memorandum, the Commission asked the ACRS to assist the staff in encouraging the industry to develop improved NDE techniques for steam generator tube inspections. The industry presentations at our meeting indicated that substantial progress is being made on the development of techniques that will provide significantly improved capabilities for detecting and sizing circumferential flaws. Not surprisingly, industry efforts are focused on a rapid resolution of the circumferential cracking problem using evolutionary improvements in eddy current technology. In addition, development is proceeding on innovative techniques such as ultrasonic guided (Lamb) waves, in situ fluorescent dye-penetrant inspections, in situ tube burst pressure testing, and combined ultrasonic and eddy current probes. Improved methods of signal processing and display are being developed to aid interpretation of NDE results. We believe modern, real time, signal processing technologies could provide great improvements in signal interpretation, defect detection, and defect sizing.

The staff and industry both recognize that the current regulatory approach to steam generator inspections discourages the development and adoption of improved NDE techniques. In the current framework, an increased detection capability leads to more plugging or repairs

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without necessarily improving safety. We believe that adoption of a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop more effective NDE techniques. Careful thought must be given to the requirements for adequate "performance demonstrations" of the NDE techniques essential for implementing a new rule. The steam generator mockup being developed by Westinghouse Electric Corporation under the Office of Nuclear Regulatory Research sponsorship may provide a useful independent regulatory check on the adequacy of NDE inspection techniques.

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

Sincerely,

T. S. Kress
Chairman

References:

1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary of the Commission, Subject: Meeting with ACRS, June 8, 1995
2. NRC Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," dated December 23, 1994
3. NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," dated April 28, 1995

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

October 13, 1995

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

Dear Chairman Jackson:

SUBJECT: NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY ON "DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS IN NUCLEAR POWER PLANTS, SAFETY AND RELIABILITY ISSUES" - PHASE 1

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 report on Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues. The NAS/NRC Committee Chairman described the results of the Phase 1 report. We also had the benefit of the documents referenced.

The objective of the Phase 1 study was to define the important safety and reliability issues concerning hardware, software, and human-machine interfaces that arise from the use of digital instrumentation and control technology in nuclear power plant operations. The report identifies eight key issues: six technical and two strategic. It notes that these issues are common to other industries where software is required for dependable operation of systems. The report succinctly presents the issues that the NAS/NRC Committee found to be important.

We agree that the issues identified in the Phase 1 report will be important considerations as digital technology is used more extensively in nuclear power plants. In the past, we have called attention to the effects of environmental stressors. The NAS/NRC Chairman stated that the NAS/NRC Committee considered, but decided not to raise this issue to the level of a "key technical issue." We continue to believe this is an important issue that the staff must address as it develops its regulatory guidance for digital systems. However, this is part of the broader issue of environmental qualification of safety-related equipment and does not need to be a key issue of the Phase 2 study.

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We have concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. We believe it is important that the SRP and other regulatory guidance benefit from the insights in the Phase 2 report.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Report dated 1995, from the Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety, Board on Energy and Environmental Systems, Commission on Engineering and Technical Systems, National Research Council, Subject: Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues - Phase 1
2. Memorandum dated December 2, 1993, from Ivan Selin, Chairman, NRC, to NRC Commissioners, Subject: Computers in Nuclear Power Plant Operations
3. Letter dated July 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems
4. Letter dated August 23, 1994, from Ivan Selin, Chairman, NRC, to T. S. Kress, Chairman, ACRS, regarding ACRS letter of July 14, 1994 on National Academy of Sciences/National Research Council Proposal for a Study and Workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety"

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

October 16, 1995

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

Dear Chairman Jackson:

SUBJECT: FATIGUE ACTION PLAN

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we completed our deliberations on the Fatigue Action Plan that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff regarding this matter and of the documents referenced.

The Fatigue Action Plan was developed to help resolve Generic Issue 166, "Adequacy of Fatigue Life of Metal Components." It was intended to address three specific issues: (1) the margin against fatigue failure of older nuclear power plants with reactor coolant pressure boundary components designed to ANSI B31.1 requirements rather than the newer ASME Code Section III, Class 1 fatigue requirements; (2) the effects of reactor coolant environments on fatigue life; and (3) the appropriate staff actions when components have cumulative usage factors (CUFs) greater than 1.

The work done on the Fatigue Action Plan by the staff and the additional work supported by the Department of Energy and the Electric Power Research Institute have shown that, even after including environmental effects, the CUFs for almost all reactor components which were originally designed to ASME Code fatigue requirements will still be less than 1. It also showed that the nuclear piping, which had been designed to the ANSI B31.1 requirements, in general has margins against fatigue failure comparable to those achieved by using the ASME Section III, Class 1, fatigue requirements. Although fatigue failures have been experienced in nuclear plants, these failures have been due to unanticipated loads and not to inadequate design margins for the anticipated cyclic loads.

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Based on a probabilistic parametric study, the staff concluded that even if fatigue cracks were initiated, rupture of reactor coolant piping as a result of fatigue crack growth would be a low-probability event. We anticipate commenting on this parametric study at a later time.

The summary of the Fatigue Action Plan provides only general guidance for the appropriate actions to be taken when the CUF is greater than 1. However, the supporting documentation suggests that the proposed nonmandatory appendix to Section XI of the ASME Code provides evaluation methods which may be acceptable to the staff. These methods provide a choice of either the traditional CUF approach or a "flaw-tolerance" approach similar to that widely used in the aerospace industry. We agree that these types of evaluations would be appropriate.

We agree with the staff that maintaining the integrity of the reactor coolant pressure boundary is an important element in defense-in-depth, and that fatigue is a potentially significant mechanism which can degrade the integrity of the pressure boundary. But, on the basis of the work done by the staff and industry, no immediate staff or licensee action is needed.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Draft Commission Paper, received August 30, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Completion of the Fatigue Action Plan (Predecisional)
2. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," published March 1995
3. SECY-94-191 dated July 26, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Fatigue Design of Metal Components
4. Staff Requirements Memorandum dated May 21, 1993, from Samuel Chilk, Secretary of NRC, to John T. Larkins, Executive Director, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards, Friday May 14, 1993

5. Letter dated August 17, 1992, from David A. Ward, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Related Branch Technical Position On Fatigue Evaluation Procedures

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

November 13, 1995

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

Dear Mr. Taylor:

SUBJECT: NUREG-0700, REVISION 1, "HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE"

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, we heard presentations by and held discussions with the NRC staff concerning the subject Design Review Guideline. We also had the benefit of the document referenced.

An outgrowth of the Three Mile Island accident was an NRC requirement that all licensees and applicants for commercial nuclear power plant operating licenses conduct detailed control room design reviews, including reviews of remote shutdown panels, to identify and correct design deficiencies related to human factors. Extensive guidelines published as NUREG-0700, "Guidelines for Control Room Design Reviews," were prepared to support these reviews.

The introduction of computer-based, human-system interface (HSI) technology into nuclear power plants prompted the development of Revision 1 to NUREG-0700. The objective of this document is to provide guidance to the NRC staff for HSI reviews of design submittals or as part of an inspection or other type of regulatory review.

The staff has developed technically defensible principles in Part 1 and a set of guidelines for HSI design reviews in Part 2. However, we are concerned that the detailed HSI design review guidance in Part 2 may discourage the approval of other, equally acceptable alternatives. Furthermore, we are concerned that the guidelines in Part 2 will become de facto regulations.

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We plan to continue our review of the overall human factors program.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Reference:
U. S. Nuclear Regulatory Commission, NUREG-0700, Revision 1, "Human-System Interface Design Review Guideline," dated January 1995

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

November 14, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.164, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS," TO RESOLVE GENERIC SAFETY ISSUE B-17

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, we reviewed the proposed final Regulatory Guide 1.164, which was developed by the staff to resolve Generic Safety Issue B-17, "Criteria for Safety-Related Operator Actions." During the meeting, we had the benefit of discussions with the NRC staff. We also had the benefit of the documents referenced.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal and accident conditions. Generic Safety Issue B-17 called for the development of time criteria for safety-related operator actions that included a methodology for determining whether or not automatic actuation would be needed to mitigate a design-basis event.

In Regulatory Guide 1.164, the staff endorses ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions." This Standard establishes criteria and simplifies the process for calculating the minimum allowable response times for manual operator actions to stabilize the plant during a design-basis event. The NRC staff proposes endorsement of this Standard to resolve Generic Safety Issue B-17.

Based on material presented by the staff, we find no technical basis for the estimates of minimum times for operator actions in ANSI/ANS-58.8-1994. Comparison of the recommended times with results from exercises done on plant simulators does not demonstrate that these times are appropriately conservative. Consequently, we do not support the staff's endorsement of ANSI/ANS-58.8-1994 in Regulatory Guide 1.164 and do not believe

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that this endorsement is the appropriate way to resolve Generic Safety Issue B-17.

The Standard does not address operator response times for advanced nuclear power plants. There is a need to consider this issue in some way for the evolutionary and passive plants.

Additional comments by ACRS Members George Apostolakis, Ivan Catton, and Robert L. Seale are presented below.

Sincerely,

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, Ivan Catton, and Robert L. Seale

In support of its recommended minimum response times, the staff relied in part on results that were produced from the Operator Reliability Experiments. We find this to be inappropriate because these experiments were not subjected to independent peer review and the staff did not have access to the actual data collected.

References:

1. Memorandum dated October 4, 1995, from M. Wayne Hodges, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Regulatory Guide 1.164, "Time Response Design Criteria for Safety-Related Operator Actions," for ACRS Review and also transmitting staff response to public comments
2. U. S. Nuclear Regulatory Commission, NUREG-0933, Supplement 06, March 1987, "A Prioritization of Generic Safety Issues," Item B-17, "Criteria for Safety-Related Operator Actions," Revision 2
3. American Nuclear Society, ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions," approved by the American National Standards Institute, Inc., August 23, 1994