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Tel. 301/415-8200 (Tuesday, July 19, 1994)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received the two attached letter-type reports from its independent Advisory Committee on Reactor Safeguards. They provide comments on:

1) protective action guidelines in emergency planning; and

2) a proposal by the National Academy of Sciences/National Research Council for a study and workshop on digital instrumentation and control systems for nuclear power plants.

In addition, the ACRS sent two letter reports to the NRC's Executive Director for Operations. They deal with areas for potential NRC staff consideration for operating nuclear power plants and future plant designs and a proposed resolution of generic safety issue 15, "Radiation Effects on Reactor Pressure Vessel Supports."

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Attachments:
As stated

July 13, 1994

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

Dear Chairman Selin:

SUBJECT: EMERGENCY PLANNING ZONES, PROTECTIVE ACTION GUIDELINES, AND THE NEW SOURCE TERMS

During the March 10, 1994 meeting with the Commissioners, the ACRS agreed to consider the implications of the results reported in the ASEA Brown-Boveri Combustion Engineering (ABB-CE) Standard Safety

Analysis Report for System 80+ design that the calculated doses for the design basis accidents (DBAs), using the new source terms and a hypothetical site, were less than protective action guidelines (PAGs) levels at the site boundary. During our 410th meeting on June 9-10, 1994, we had the benefit of a staff presentation on the use of PAGs in emergency planning. We also had the benefit of the referenced documents.

Calculated doses associated with the DBA prescription are sensitive to parameters associated with the DBA specifications, the containment design, and the site characteristics. These parameters include, for example, the source term itself (amount, timing, and chemical form), the effectiveness of engineered and natural aerosol mitigation processes (e.g., sprays and containment dimensions), containment volume and leak rate, the associated DBA pressure source, and specified meteorological conditions.

The items that appear to be major contributors to the low dose values calculated for System 80+ are:

- the large volume of the containment,
- an effective spray system design,
- an annular containment design that routes leakage through a filtered vent,
- the new specification for the source term contained in draft NUREG-1465 (particularly the timing), and
- the use of "medium" meteorological conditions as taken from the EPRI Utility Requirements Document for a hypothetical site instead of "worst-case" conditions.

The implication of the low value of the calculated DBA dose at the site boundary is that it points to a need to revisit the technical basis and rationale that underlie the present regulatory guidance on emergency planning - particularly with respect to the extent of Emergency Planning Zones (EPZs). This is an opportunity to develop a trial application of the concept of risk-based regulations.

The existing regulations require that emergency response plans be established and the guidance calls for including provisions for sheltering and/or evacuating within a 10-mile radius (i.e., plume exposure EPZ) around the reactor site in the event that doses anywhere in that region during an accident in progress are <u>projected</u> to exceed the PAGs. In addition, a 50-mile ingestion pathway zone is called for such that protective measures are available in the event that projected doses exceed additional PAG values in that zone.

The rationale for these requirements seems to be defined in NUREG-0654, from which we cite the following:

> "... it would be unlikely that any protective action for the plume exposure pathway would be required beyond the plume exposure EPZ."
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> "... the likelihood of exceeding ingestion pathway protective action guide levels at 50 miles is comparable to the likelihood of exceeding plume exposure pathway protective action guide levels at 10 miles."
>
> "Projected doses from most core melt sequences would not exceed PAGs outside the [10-mile] EPZ."
>
> "For the worst core melt sequences, immediate life threatening doses would generally not occur outside the [10-mile] EPZ."

This is a good example of the type of regulatory basis that has concerned the ACRS for years. It has the "right-sounding" words but is lacking in real substance and is inflexible for new designs. In particular, it has only a loose risk basis rooted primarily in the results from WASH-1400, is specific only for contemporary LWRs, and uses qualifiers such as "unlikely," "likelihood," "most," and "generally." We believe the regulations related to emergency planning deserve better.

We believe the current regulatory extent of the EPZs as applied to existing nuclear plants implies an underlying level of "accepted risk." If a comparable risk basis were to be applied to advanced plants, then the associated resulting EPZs would be expected to be smaller, possibly shrinking to the size of the site boundary.

The Commission, in the July 30, 1993 SRM, directed "... the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplifications of existing emergency planning requirements." We support this directive from the Commission and note that, as part of the draft PRA implementation plan, the staff intends to proceed with efforts in that direction. We recommend that, as part of this effort, the staff be directed to develop firm risk-based criteria for EPZs for use with advanced plant designs. We believe developing such criteria would first require developing answers to the following questions:

- What level of risk is being "accepted" for currently operating LWRs with their existing EPZs?
- Is this level of "accepted" risk appropriate? If not, what should it be?
- For the advanced plant designs, what would be the size of the EPZs based on a level of risk comparable to the "accepted" value? What are the implications of this result?

We recognize that developing criteria based on "acceptable risk" would be conceptually as difficult as was development of the Safety Goal criteria. We also recognize that defense-in-depth might be a sufficient regulatory basis for the present extent of EPZs. Nevertheless, we believe that now is the appropriate time, and that the guidance on EPZs is the appropriate subject, for a trial effort on risk-based regulation to begin.

Sincerely,

T. S. Kress, Chairman
Advisory Committee on
Reactor Safeguards

References:
1. Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. U.S. Nuclear Regulatory Commission, NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978
3. U.S. Nuclear Regulatory Commission, NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," February 1980
4. Staff Requirements Memorandum dated July 30, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements

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July 14, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY AND WORKSHOP ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we discussed the proposal by the National Academy of Sciences/National Research Council (NAS/NRC) for a study and workshop on the "Application of Digital

Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety." During our review, we had the benefit of discussions with representatives of the NRC staff and the NAS/NRC. We also had the benefit of the documents referenced. This report is in response to a Commission request in the March 18, 1994 Staff Requirements Memorandum.

The proposal focuses primarily on hardware and software issues that arise from the introduction of digital instrumentation and control (I&C) technology in nuclear power plants. Human factors considerations appear to be limited to human-machine interface issues related directly to digital technology. We believe this balance in emphasis is proper. The issues associated with hardware and software are very broad and any significant diversion of effort from these issues is undesirable. In addition, we believe that the staff's Human Factors Engineering Program Review Model and the acceptance criteria used for evolutionary reactors provide reasonable regulatory guidance for human factors issues. The current need is for a corresponding regulatory framework for hardware and software issues associated with digital I&C technology.

We believe the NAS/NRC study panel findings will assist the Commission in providing necessary guidance to the staff for the development of a regulatory framework for digital I&C. While the staff and the ACRS have identified a number of concerns that are believed to be significant, the ACRS strongly urges that the study panel be permitted to select the issues to be considered.

We expect that the NAS/NRC study will make use of knowledge that has been developed in other industries with digital system experience. We are particularly interested in the state-of-the-art of the development of software specifications, verification and validation of software, the potential vulnerabilities of hardware over the spectrum of adverse environments which can occur in nuclear power plants, and the prediction of reliability (including common-mode failure).

We recommend that the staff identify in the background papers provided to the NAS/NRC study panel those applicable NRC regulations, IEEE standards, Electric Power Research Institute Utility Requirements, and vendor information that pertain to safety-related digital I&C system development.

We understand that a visit to the NRC Technical Training Center simulators is planned. It may be more useful for study panel members to visit a nuclear plant digital system vendor to observe developmental mock-ups and to discuss nuclear power plant digital I&C designs. Consideration should also be given to visiting an operating plant that employs digital control and protection systems.

We look forward to meeting with members of the study panel during the course of the study.

Sincerely,

T. S. Kress, Chairman
Advisory Committee on
Reactor Safeguards

<u>References</u>:

1. Memorandum dated March 18, 1994, from Samual J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. Memorandum dated March 1, 1994, from James M. Taylor, Executive Director for Operations, NRC, for The Commission, Subject: Nuclear Safety Research Review Committee Report Dated January 14, 1994
3. Memorandum dated May 3, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, Subject: Staff Response to Nuclear Safety Research Review Committee Reports Dated January 14 and February 16, 1994
4. ACRS Letter Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations
5. ACRS Letter Report dated November 16, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

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July 13, 1994

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

Dear Mr. Taylor:

SUBJECT: SOME AREAS FOR POTENTIAL STAFF CONSIDERATION FOR OPERATING NUCLEAR POWER PLANTS AND THE REVIEW OF FUTURE PLANT DESIGNS RESULTING FROM THE ACRS REVIEW OF THE EVOLUTIONARY LIGHT WATER REACTORS

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we completed our discussion related to the results of our recent reviews of the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) (Reference 1) and the ASEA Brown-Boveri Combustion Engineering (ABB-CE) System 80+ (Reference 2) applications for design certification from the perspective of potential areas for staff action for operating nuclear power plants and the review of future plant designs. These reviews provided us with an opportunity to consider present regulatory practices and procedures vis-a-vis the "state-of-the-art" design requirements for these evolutionary light water reactors (ELWRs).

The following are some issues that we believe the staff should address as Generic Issues, as Technical Specification Improvement Program issues, as revisions to the Standard Review Plan, or as additional research needs.

1. <u>Turbine Inspection Requirements</u> - In the course of reviewing the potential for turbine rotor failure related to the ABWR and System 80+ designs, we learned that the staff has not prepared an appropriate set of preoperational and inservice inspection, evaluation and acceptance requirements for turbine rotor, other than those employing shrunk-on disks.

   Some current licensees have replaced, or are planning to replace, shrunk-on disk rotors with rotors of a different design. We believe that the staff should develop appropriate positions for the various designs on a priority basis.

2. <u>Technical Specification Requirements for Onsite Power Sources</u> - In our letter to you dated February 17, 1994, concerning three issues relating to the 10 CFR Part 52 design certification process for ALWRs, we recommended that the staff resolve the matter of credit for ELWR alternate AC sources when 1E emergency diesel generators are out of service during power operation. We suggested that Technical Specification requirements for such onsite power sources be based on appropriate probabilistic considerations. Subsequently, ABB-CE requested such credit for System 80+ and the staff has granted an allowable outage time for a 1E emergency diesel generator of up to 14 days when the combustion turbine-generator is available. We now recommend that the staff expand this concept to include operating nuclear power plants.

   It is our understanding that Technical Specification requirements for onsite power sources will be incorporated into the Shutdown and Low Power Operations Rule.

3. <u>Reactor Water Cleanup System Safety</u> - The Reactor Water Cleanup (RWCU) System is of safety concern for boiling water reactor plants because it is a high-energy, non-safety system, portions of which may be located inside of the secondary containment. The secondary containment also houses numerous engineered safety features and the Fuel Pool Cooling System. For operating plants, the RWCU System supply line from the reactor vessel is usually a 6-inch pipe. A rupture of this pipe inside of the secondary containment results in a loss of reactor coolant which may create a serious environmental

disruption throughout the secondary containment before it can be isolated.

An ACRS staff report (Reference 3) identified a number of safety-related deficiencies in a similar system for the ABWR. Subsequently, GENE developed a requirement for environmental qualification of all safety-related components and the Fuel Pool Cooling System inside of the secondary containment. The qualification was based mostly on the adverse atmosphere created before complete closure of the isolation valves following a supply line pipe break. Generally, operating plants do not provide a comparable level of environmental qualification.

Another GENE change was the addition of a second isolation valve in the supply line inside of the primary containment. This valve isolates the reactor vessel from the supply line pipe break in the event that isolation is not achieved by closing the two primary containment isolation valves under blowdown flow conditions. The added valve is not capable of blowdown isolation. It is closed by manual actuation after the blowdown is completed, thereby achieving reactor vessel isolation and interruption of any prolonged release of Emergency Core Cooling System (ECCS) water to the break which is outside of primary containment. Operating plants may not have a similar capability. We recommend that this issue be investigated for operating BWRs.

4. Review of Chilled-Water Systems - A number of operating plants use large Chilled-Water Systems to provide essential environmental cooling. Because there is no Standard Review Plan (SRP) for these systems, the staff has used other guidance such as SRP 9.2.2 (Reactor Auxiliary Cooling Water Systems) when evaluating the safety of such systems. However, this guidance is not appropriate for the evaluation of refrigeration systems.

   In determining plant safety, the NRC staff needs to evaluate the performance of Chilled-Water Systems under various accident heat loads and during loss-of-offsite-power events, and to consider the ability of such systems to restart and function after tripping or after a prolonged station blackout. We urge that the staff develop better guidance and positions with which to enhance the scope and quality of its plant reviews of Chilled-Water Systems.

5. <u>Filters or Water Separators for the Hardened Vents Installed on Operating BWR Containments</u> - A great deal of analysis was done to demonstrate that the ABWR Containment Overpressure Protection System is adequate without filters or water separators. We are not aware that such an analysis has been done for those operating BWRs with hardened vents. We believe their need for filters or water separators should be reevaluated.

6. <u>Fuel-Coolant Interactions</u> - We are concerned that the safety case with respect to fuel-coolant interactions is based mostly on arguments of low probability of occurrence. It concerns us that neither the industry nor the NRC staff is able to predict limits to the energetics (below purely thermodynamic limits) based on either first principles or sufficient empirical evidence. We believe additional research is needed on this issue.

7. <u>Adequacy and Use of PRA</u> - We are concerned that there are no clear regulatory criteria for what constitutes an acceptable PRA. By accepting the PRAs which have already been submitted, the staff is essentially establishing the regulatory criteria by precedent rather than by promulgating specific requirements. We believe consideration should be given to establishing minimum requirements for PRAs.

Sincerely,

T. S. Kress, Chairman
Advisory Committee on
Reactor Safeguards

<u>References</u>:

1. ACRS Report dated April 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on Safety Aspects of the General Electric Nuclear Energy Application for Certification of the Advanced Boiling Water Reactor Design
2. ACRS Report dated May 11, 1994, from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on the Safety Aspects of the ASEA Brown Boveri-Combustion Engineering Application for Certification of the System 80+ Standard Plant Design
3. Advisory Committee on Reactor Safeguards Report by S. E. Mays and M. E. Stella, "ABWR Reactor Water Cleanup System Review," July 30, 1992

July 13, 1994

Mr. James M. Taylor

Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 15, "RADIATION EFFECTS ON REACTOR PRESSURE VESSEL SUPPORTS"

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we reviewed the NRC staff's proposed resolution of Generic Safety Issue 15 (GSI-15), "Radiation Effects on Reactor Pressure Vessel Supports." During this meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We have no objection to the NRC staff proposal to resolve GSI-15 by issuing an Information Notice and providing a related NUREG report to all licensees.

Dr. T. S. Kress and Dr. W. J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

W. J. Lindblad
Vice-Chairman, ACRS

<u>References</u>:

1. Memorandum dated June 22, 1994, from J. A. Murphy, RES, for J. T. Larkins, ACRS, transmitting the following documents:

   - Memorandum (revision as of 5/2/94) from Eric S. Beckjord, RES, for James M. Taylor, EDO, Subject: Resolution of Generic Safety Issue 15, "Radiation Effects on Reactor Vessel Supports"

   - NRC Information Notice 94-XX (Draft) dated April XX, 1994, Subject: Generic Safety Issue 15 Resolution - Radiation Effects on Reactor Pressure Vessel Supports

   - NUREG-XXXX, Draft dated 6/22/94, Subject: Radiation Effects on Reactor Pressure Vessel Supports

   - Regulatory Analysis (Undated Draft), Subject: Resolution of Generic Safety Issue No. 15, "Radiation Effects on Reactor Vessel Supports"

   - U. S. Nuclear Regulatory Commission, NUREG/CR-6117, ORNL/TM-12484, Subject: Neutron Spectra at Different

High Flux Isotope Reactor (HFIR) Pressure Vessel Surveillance Locations, December 1993

2. ACRS report dated July 15, 1987, from William Kerr, ACRS Chairman, to Victor Stello, Jr., Executive Director for Operations, Subject: ACRS Comments on the Embrittlement of Structural Steel