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ABSTRACT

This compilation contains 30 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1994. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U. S. Library of Congress. The reports are categorized by the most appropriate generic subject area and by chronological order within subject area.

PREFACE

The enclosed reports represent the recommendations and comments of the U. S. Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards during calendar year 1994. NUREG-1125 is published annually. Previous issues are as follows:

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1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
13	Calendar Year 1991
14	Calendar Year 1992
15	Calendar Year 1993

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

December 19, 1994

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

Dear Chairman Selin:

SUBJECT: LOSS OF SPENT FUEL POOL COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT AT THE SUSQUEHANNA STEAM ELECTRIC STATION

During the 416th meeting of the Advisory Committee on Reactor Safeguards, December 8-10, 1994, we discussed the NRC staff Draft Safety Evaluation Report (DSER) dealing with the potential for loss of spent fuel pool cooling following a loss-of-coolant accident (LOCA) at the Pennsylvania Power and Light (PP&L) Company's Susquehanna Steam Electric Station Units 1 and 2. During the meeting, we had the benefit of discussions with representatives of the NRC staff, PP&L, and the individuals who brought this matter to the attention of the NRC on November 27, 1992, through a 10 CFR Part 21 notification. We also had the benefit of the documents referenced. We considered this matter previously during our May 5-7, 1994 meeting.

The 10 CFR Part 21 notification described the individuals' concerns with: (1) the ability of Susquehanna to provide adequate cooling of the spent fuel storage pool following various design-basis LOCAs; (2) the potential causes and consequences of failure to cool the spent fuel storage pool; and (3) numerous regulatory issues regarding potential design deficiencies.

The primary concern raised by the two individuals was a postulated failure to cool the spent fuel storage pool following a design-basis LOCA or a LOCA with a loss of offsite power (LOOP). They posited that a design-basis LOCA would result in the failure of the nonsafety-related spent fuel pool cooling system. They further posited that a design-basis LOCA results in the development of a TID 14844-like radiological source term inside the reactor building that would prevent operators from entering the building and restoring cooling to the spent fuel pool. The individuals further postulated that, upon boiling in the pool, vapor would be transported throughout the reactor building by the ventilation systems and would eventually cause the failure of safety-related systems needed to mitigate the LOCA. The ultimate consequences of

these boiling scenarios include severe core damage, failure of the stored spent fuel, and loss of primary and secondary containment.

The DSER, which stands separate from the staff's regulatory compliance evaluation, includes a review of certain specific aspects of the Susquehanna facility design and a deterministic examination of some of the physical phenomena involved. The evaluation also includes a probabilistic analysis of postulated event sequences involving loss of the spent fuel storage pool cooling.

In our review of this matter, we were looking for answers to three questions:

1. Is Susquehanna now operating without undue risk to the health and safety of the public?
2. Was Susquehanna operating in an unsafe condition prior to modifications and procedural changes that have been made?
3. Are there generic implications of undue risk at other operating plants?

Additionally, we have an interest in whether or not the postulated pool boiling sequences should have been part of the design-basis accident and, thus, part of the licensing basis for Susquehanna. Our interest here stems from our concerns about coherence in the regulatory process and about ill-advised actions that can create burdens on licensees without providing a corresponding increase in safety.

Clearly, the appropriate approach to answering the first question is to conduct a limited probabilistic risk assessment (PRA) for the plant as now configured, focusing on the LOCA sequences that can lead to spent fuel pool boiling. The staff has done this and found that the core-damage frequency (CDF) is less than $1 \times 10^{-6}/\text{yr}$. This clearly indicates that the plant is not at undue risk from these particular sequences.

The appropriate approach to answering the second question is to repeat the limited PRA but with the plant in the as-found configuration before any modifications. The staff has conducted this study and found that the risk was similarly low, with a CDF of $4 \times 10^{-6}/\text{yr}$.

Our opinion on this issue rests on how well we think these PRAs were done and whether or not the results are credible. Since we did not review these PRAs in any detail, we are unable at this time to make a judgment as to their quality. Because the safety case rests primarily on the validity of the results of these PRAs, we recommend that the PRAs and their associated uncertainty analyses

be given a thorough review. The reviewers should pay particular attention to the treatment given the environmental effects brought about by LOCAs, including interfacing system LOCAs. This area of PRA could use additional research by NRC.

We cannot judge the generic implications. The low risk for the "as-found" configuration (before modifications), indicated by the PRA result, indicates to us that spent fuel pool boiling is not likely to be of concern as a risk-contributor at other plants. Nevertheless, we think it appropriate that NRC issue a generic notification to all licensees describing this particular issue and requesting a review of plant vulnerability to spent fuel pool boiling. This could be an adjunct to the Individual Plant Examination (IPE) process.

With respect to the licensing-basis issue, we have the following opinion. If the PRA result indicating very low risk is correct, then it would be inappropriate at this time to consider augmenting the Susquehanna licensing basis with the postulated pool-boiling sequences.

Sincerely,

T. S. Kress
Chairman

References:

1. Letter dated October 24, 1994, from Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to J. T. Larkins, Executive Director, ACRS, Subject: 409th ACRS Meeting Followup Matters and transmitting Draft Safety Evaluation Report
2. Letter dated May 16, 1994, from D. Lochbaum and D. Prevatte, Members of Public, to J. T. Larkins, Executive Director, ACRS, Subject: Susquehanna Steam Electric Station Units 1 and 2 Loss of Spent Fuel Pool Cooling Licensing Basis

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

January 14, 1994

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

Dear Chairman Selin:

SUBJECT: FINAL REPORT ON THE USE OF THE DESIGN ACCEPTANCE CRITERIA
PROCESS IN THE CERTIFICATION OF THE GENERAL ELECTRIC
NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 405th meeting of the Advisory Committee on Reactor Safeguards, January 6-7, 1994, we completed our review of the Design Acceptance Criteria (DAC) to be included in the Certified Design Material (CDM) for the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR). The four subject areas addressed by DAC are Human Factors Engineering, Radiation Protection, Piping Design, and Instrumentation and Control.

Our Ad Hoc Subcommittee on DAC, in a joint meeting on November 2, 1993, with the Computers in Nuclear Power Plant Operations Subcommittee, reviewed Chapter 7, "Instrumentation and Control Systems," of the GENE Standard Safety Analysis Report (SSAR), the NRC staff Final Safety Evaluation Report (FSER) for this Chapter, and the related DAC. This DAC was further discussed during our November 4-6, 1993 meeting. Our ABWR Subcommittee, during its meeting of November 17, 1993, reviewed the human factors aspects of Chapter 13, "Conduct of Operations," and Chapter 18, "Human Factors Engineering," of the GENE SSAR, the NRC staff FSER for these Chapters and the related DAC for Human Factors Engineering. The DACs on Radiation Protection and Piping Design were discussed during our December 9-11, 1993 meeting. In each of these meetings, we had the benefit of discussions with representatives of the NRC staff and GENE. We also had the benefit of the documents referenced.

In addition to the meetings described above, both ACRS and its Ad Hoc Subcommittee on DAC (which was established to review the DAC process as requested by the Commission in its April 1, 1992 Staff Requirements Memorandum) met on a number of occasions to consider the overall DAC process as it was evolving. We provided two interim reports during this period. With this report, we believe that the Ad Hoc Subcommittee on DAC has now completed its assignment.

BACKGROUND

Since our last report, considerable effort has been expended by the NRC staff, GENE, NUMARC, and interested industry participants in the development of the Tier 1 CDM for the ABWR. As described in the GENE CDM submittal of December 7, 1993, the Tier 1 CDM relevant to the four subject areas that use the DAC process is contained in Section 3.0 "Additional CDM." This section consists of those aspects of the certified design that do not lend themselves to the system-by-system coverage provided in Section 2.0 of the CDM for individual plant systems. Each of the four DAC CDM sections consists of a Design Description and associated Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). Certain elements of these ITAAC are designated as DAC because they describe the design process to be used in implementing the design commitments stated in the Design Description. This is in contrast to the general case in which ITAAC will be used to confirm that the as-built plant systems have the design characteristics stated in the Design Description. Both the CDM and the associated Tier 2 material constitute the complete set of requirements for the certified design.

RECOMMENDATIONS AND COMMENTS

With respect to the material in Section 3.0 "Additional CDM" covering the four subject areas historically referred to as DAC, we are generally satisfied that it provides a reasonable basis for the staff final safety determination needed to support Final Design Approval. Our comments on each of these CDM are as follows:

Section 3.1 - Human Factors Engineering (HFE)

This section imposes Tier 1 requirements on the Combined Operating License (COL) holder with respect to the implementation of the human-system interface (HSI) for certified design. All six elements of ITAAC associated with this CDM have been designated as DAC by the staff and GENE.

Our review of HSI covered Chapter 18 of the FSER and the "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors," both dated December 1993. The latter document provides the technical basis for the staff review of the HFE design process proposed for certification. It also specifies the acceptance criteria by which the staff will evaluate the HFE program elements proposed by an applicant. We commend the staff for the development of this document. It provides much needed guidance to applicants on the staff expectations with regard to HFE for evolutionary reactors.

The HSI scope is limited to the main control room and the remote shutdown system. We commented, in our report of June 16, 1992, that the scope of the DAC then under development should be expanded

to include "... transmission switchyard work stations, because of the importance of offsite power to the safety of nuclear power plant operations" and "... incorporation of human factors principles in the design of local panels where instrumentation and controls important to safety are located." Although not included in this section of the CDM, we believe that these issues have been appropriately addressed elsewhere in the CDM.

Section 3.2 - Radiation Protection

This section imposes Tier 1 requirements on the COL holder with respect to the design of radiological shielding and ventilation systems. The scope of this section includes the design of these features for the Reactor Building, Turbine Building, Control Building, Service Building, and Radwaste Building. All six elements of ITAAC associated with this section have been designated as DAC by the staff and GENE.

The Design Description requires that the plant shielding design permit operators to perform required safety functions in "vital areas" of the plant under "accident conditions." The definition of "vital areas" in the Design Description differs from that in 10 CFR 73.2. We believe that other terminology should be used in this Design Description to avoid confusion with the definition used by the nuclear power plant security community.

ITAAC 3 of Table 3.2a contains the design commitment that "the plant shielding design shall permit plant personnel to perform required safety functions ... under accident conditions," and defines the accident radiation source term to be used for the shielding design. We agree that this source term is appropriate for this purpose.

Acceptance Criteria 1.a, b, and c of Table 3.2b distinguish, for purposes of ventilation system design, among "normally occupied rooms," "rooms that require infrequent access," and "rooms that seldom require access." The distinction between 1.b and 1.c is not obvious and should be more sharply drawn.

Section 3.3 - Piping Design

This section imposes Tier 1 requirements on the COL holder with respect to: (1) the design of nuclear safety-related piping systems and certain non-nuclear safety-related piping systems; (2) the analysis of the dynamic effects associated with postulated high energy pipe breaks on structures, systems, and components that are required to be functional during and following a safe shutdown earthquake; and (3) the reconciliation analysis of the as-built piping against the piping design. All three elements of this ITAAC have been designated as DAC by the staff and GENE.

The scope of this section is spelled out in the Design Description. There are, however, a number of additional aspects of piping design and analysis important to nuclear power plant safety which are not covered by this section. These have been discussed in detail with the staff and GENE on a number of occasions. We have been told that these piping design and analysis issues will be included elsewhere in the CDM. We will continue to follow this matter until we are satisfied that these issues have been properly addressed.

Section 3.4 - Instrumentation and Control

This section imposes Tier 1 requirements on the COL holder with respect to: (1) the configuration of safety-related digital instrumentation and control (I&C) equipment encompassed by the Safety System Logic and Control (SSLC); (2) the hardware and software development process used in the design, testing, and installation of I&C equipment; and (3) the diverse features included in I&C system design to provide backup support for postulated worst-case common-mode failures of SSLC. ITAAC 7 through 11 have been designated as DAC by the staff and GENE.

We would have preferred that the staff had based its review and acceptance of this section, the related Section 2.0, and SSAR Chapter 7 on a documented review model and specific acceptance criteria, as was done in the case for the Human Factors Engineering section discussed above. The staff has not yet formulated an identifiable set of criteria which must be met by digital I&C systems. In the FSER, reference is made to a menagerie of NRC regulations and regulatory guides, to a set of industry standards, and to several NRC publications which provide the basis for the staff conclusions concerning the process being followed by GENE. However, an examination of these indicates that most were developed before any significant application of digital technology to reactor safety systems, that only a few are relevant to many of the staff concerns, and that several are obsolescent if not obsolete.

We continue to recommend that the staff produce, on an expedited basis, a soundly conceived Standard Review Plan for digital I&C systems for both ALWRs and operating plant backfits.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. GE Nuclear Energy, "ABWR Certified Design Material," Volumes 1 and 2, December 7, 1993

2. GE Nuclear Energy, "ABWR Standard Safety Analysis Report," September 1993
3. Staff Requirements Memorandum from Samuel J. Chilk, Secretary of the Commission, to David A. Ward, ACRS Chairman, dated April 1, 1992, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992
4. NRC staff Final Safety Evaluation Report for the General Electric Nuclear Energy Advanced Boiling Water Reactor, December 1993
5. NRC staff Final Safety Evaluation Report for the General Electric Nuclear Energy Advanced Boiling Water Reactor, "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors" (Appendix 18A), December 1993
6. ACRS report dated June 16, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Interim Report on the Use of Design Acceptance Criteria in the Certification of the GE Nuclear Energy Advanced Boiling Water Reactor Design
7. ACRS report dated October 16, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Second Interim Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design

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

March 15, 1994

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

Dear Chairman Selin:

SUBJECT: DRAFT COMMISSION PAPER ON SOURCE TERM RELATED TECHNICAL
AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND
PASSIVE LIGHT WATER REACTOR DESIGNS

During the 406th and 407th meetings of the Advisory Committee on Reactor Safeguards, February 10-11 and March 10-12, 1994, respectively, we discussed the draft Commission paper on source term related technical and licensing issues pertaining to evolutionary and passive light water reactor (LWR) designs. During these meetings, we had the benefit of discussions with representatives of the NRC staff and industry. We also had the benefit of the documents referenced.

Separate source terms are provided for BWRs and PWRs. The source terms consist of the fraction of the equilibrium core inventory of fission products released into containment, the timing of this release, and the chemical form of the fission product iodine. In the past, such source terms have been specified in Regulatory Guides 1.3 and 1.4 to provide guidance on appropriate values to use in the site suitability analyses that are required by 10 CFR Part 100, and in conjunction with the other design basis accidents (DBAs) in Chapter 15 of the Standard Review Plan. The DBA source terms should not be confused with the plant and sequence specific source terms that are mechanistically derived and used in PRAs and other severe accident analyses. The specifications that are presently in Regulatory Guides 1.3 and 1.4 consist of 100 percent of the noble gases and 25 percent of the iodine (91 percent as elemental iodine, 5 percent as particulate iodine, and 4 percent as organic iodine). For site suitability analyses, these specifications have been used along with a thermal hydraulic specification. These analyses require that a peak containment pressure be calculated for a double-ended break of the largest primary system piping and be applied for 24 hours after which it is to be reduced to half that value.

The 10 CFR Part 100 specifications of the source term have always been viewed as being somewhat arbitrary, but conservative. The proposed revised source terms are intended to remove some of the arbitrariness of the present values and to make them more realistic. As part of the overall process of decoupling site suitability decisions from reactor design, the revised source term and the dose criteria provisions are to be removed from 10 CFR Part 100 and put into 10 CFR Part 50 where they would apply only to design features. The revised source terms are based on values developed in NUREG-1150 for the "in-vessel" release phase associated with severe accidents.

In the draft Commission paper, the staff describes the proposed revised source terms and proposed uses for reviews and assessments of evolutionary and passive LWR designs. The paper discusses positions taken by the staff on source term issues for evolutionary and passive LWR designs (identified in SECY-90-016 and SECY-93-087). The staff believes these positions will provide a basis for closing these issues with respect to design certification reviews and the EPRI Utility Requirements Documents.

We generally agree with the positions taken by the staff on the issues and agree with the principle that the source terms for DBAs should be made more realistic. Realistic source terms should result in more appropriate designs (e.g., engineered safety features, source term mitigation features, sampling and measurement devices, and containment integrity). We believe the changes can lead to increased coherence in the associated regulations and their application. As in all responses to the accumulation of new knowledge, such proposed changes in the regulations, whether toward enhancement or relaxation, or whether applied to existing plants or to future plants, should be assessed for their overall effect on risk. We also have the following concern about the revised source term specifications.

We think the realistic specification of the thermal hydraulics and production of nonradioactive aerosols associated with the DBAs is as important as the specification of the source term itself. These conditions can strongly influence the behavior of radioactive aerosols in containment. Additional consideration should be given to developing Commission guidance on the thermal hydraulic conditions and nonradioactive aerosol generation to be coupled with the source terms for the various DBAs.

We continue to recommend that the General Design Criteria for containment volume and strength for future ALWRs incorporate the spectrum of severe accident challenges described in our report of

May 17, 1991. The containment should represent a defense-in-depth feature that is not limited to design basis accidents.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated January 6, 1994, from Dennis M. Crutchfield, NRC Office of Nuclear Reactor Regulation, for John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Commission Paper on Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs
2. Memorandum dated February 10, 1994, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Draft Commission Paper, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs"
3. SECY-93-087, Memorandum dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
4. SECY-90-016, Memorandum dated January 12, 1990, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
5. NUREG-1150, Volumes 1 and 2, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power plants," December 1990
6. Report dated May 17, 1991, from David A. Ward, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Proposed Criteria to Accommodate Severe Accidents in Containment Design

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

April 14, 1994

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

Dear Chairman Selin:

SUBJECT: REPORT ON SAFETY ASPECTS OF THE GENERAL ELECTRIC NUCLEAR
ENERGY APPLICATION FOR CERTIFICATION OF THE ADVANCED
BOILING WATER REACTOR DESIGN

During the 408th meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 1994, we completed our review of the General Electric Nuclear Energy (GENE) application for certification of its U.S. version of the Advanced Boiling Water Reactor (ABWR) standard design. This final report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review we had the benefit of discussions with representatives of GENE and the NRC staff. We also had the benefit of the documents referenced.

ABWR Application

The U.S. version of the ABWR standard design utilizes a significant portion of the detailed design information developed jointly by GENE, Hitachi, and Toshiba for the international version which is being built in Japan. The application for certification of the U.S. version was filed by GENE in September 1987 under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). The application was docketed in February 1988. In December 1991, GENE requested that the application be considered under 10 CFR 52.45. This request was made effective in March 1992.

The application is based on the ABWR Standard Safety Analysis Report (SSAR), which was submitted in modular form between September 1987 and March 1989. Since then it has been amended frequently, the last submittal for our review was Amendment 34 in March 1994. The application also includes the ABWR Certified Design Material (CDM). The CDM contains the design information from the SSAR that will become a part of the design certification rule. The CDM has been revised, the last submittal that we received was Rev. 2 in December 1993.

ABWR Design Description

The ABWR is a forced circulation boiling water reactor with a rated power of 3926 MWt. The reactor core consists of 872 8x8 fuel assemblies and 205 control rods. The reactor utilizes internal recirculation pumps and fine-motion control rod drives. It is located inside a steel-lined reinforced concrete pressure suppression containment which is enclosed by a reinforced concrete secondary containment, both of which are located in the Reactor Building. The Reactor Building also houses a standby gas treatment system, refueling area, main steam pipe tunnel, and essential systems for emergency core cooling, AC power (including diesel generators), and environmental conditioning.

The Control Building is located between the Reactor Building and the Turbine Building. The Control Building houses a continuation of the main steam pipe tunnel, the main control room, a computer facility, and essential systems for DC power, environmental conditioning, and cooling water. During emergencies, technical support is provided by the Technical Support and Operational Support Centers, which are located in the Service Building, which is immediately adjacent to the Control Building.

The Turbine Building houses equipment for power generation. Steam is supplied to an 1800 rpm turbine-generator which is oriented to minimize damage to safety-related equipment should a turbine failure occur. The Turbine Building also houses systems and equipment that provide various nonessential services for the plant. These include the standby combustion-gas-turbine generator, house boiler, air compressors, and systems for AC and DC power and environmental conditioning.

The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant. An underground pipe tunnel connects the Turbine and Reactor Buildings to the Radwaste Building.

The ABWR design includes a number of features that we believe will enhance safety relative to past BWR designs. Some of these features resulted from the use of PRA methodology by GENE in evaluating the ABWR design as it progressed.

- The use of reactor internal pumps removes the large reactor recirculation piping and connections to the reactor vessel, thereby reducing the size of the largest loss-of-coolant accident (LOCA).
- The use of a fine-motion control rod drive arrangement removes the scram discharge volume and associated piping, provides two reliable means for inserting the rods, and is intended to eliminate the rod drop and rod ejection accidents.

- The Emergency Core Cooling System and supporting auxiliaries are arranged into three physically separated electrical and mechanical divisions, only one of which is needed for handling transients and virtually all accidents.
- A combustion-gas-turbine generator is provided for enhanced on-site AC power capability.
- An AC-independent reactor water addition feature, a depressurization system, lower drywell flooder, cavity floor spreading area, sacrificial layer of basaltic concrete, and containment overpressure protection system are provided to mitigate severe accidents.
- The greatly increased application of digital control systems offers the potential for improved operator interface with the plant and the reliability of control and protection systems. In addition, the use of digital multiplexers and fiber optics reduces the amount of cabling in the plant thereby reducing the fire hazard.
- The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. This, in combination with improved material specifications, reduces concern for reactor vessel integrity.

Chronology of ACRS Review

Our review of the ABWR application commenced after it was docketed in February 1988. The NRC staff issued a Draft Safety Evaluation Report (DSER) on the first module of the SSAR in August 1989 (Ref. 2). We reviewed this draft and reported our findings in November (Ref. 3). At that time we questioned, in particular, the adequacy of the level of design detail available for review and recommended that the staff revisit the issue of what constitutes an "essentially complete" design.

Subsequent to November 1989, our review activities focused on several ABWR-related design concerns including Control Building flooding, physical separation, environmental protection of sensitive equipment, performance of essential chilled water systems, use of leak-before-break methodology, use of integral low-pressure turbine rotors, and the capability of the floor area beneath the reactor vessel to cope with severe accidents. These preliminary concerns were brought to the attention of the NRC staff in our July 1991 report (Ref. 4).

During 1991 the DSER was completed by the NRC staff in the form of six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These papers generally covered most sections of the SSAR through the first eighteen amendments, but contained numerous open items. We

reported our findings in April 1992 (Ref. 5). In this report, we reconfirmed the preliminary concerns expressed in our July 1991 report and added several more including adequacy of the PRA, containment hydrodynamic loads, Reactor Water Cleanup System safety implications, plant design life and aging management, station grounding and surge protection, and corrosion control for structures.

In October 1992, the NRC staff issued a Draft Final Safety Evaluation Report (DFSER) (Ref. 6) covering the entire SSAR through Amendment 20. This draft superseded the six SECY papers. The final version of the staff safety evaluation report which we reviewed was the "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," dated December 1993 (Ref. 7). This copy covered the NRC staff review of SSAR information through about Amendment 32. Additional changes, including those which reflect Amendments 33 and 34, were reviewed by us as page changes to Reference 7.

Between February 1988 when the ABWR application was docketed and April 1992 when we issued our report on the DSER, our ABWR subcommittee held numerous meetings to review the SSAR and the NRC staff safety evaluations. During this same period, our subcommittee on Improved Light Water Reactors held several meetings to review the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) and associated NRC staff safety evaluations for the Advanced Light-Water Reactor (ALWR) evolutionary plant. (The EPRI URD prescribes ALWR design requirements from the utility industry perspective.) Meetings were also held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, and Severe Accidents. These subcommittees reviewed a number of specialized aspects of the proposed ABWR design including those related to fire, digital control and protection systems, human factors, and severe accidents.

Between April 1992 and today, our ABWR subcommittee held additional meetings to review design features proposed beyond Amendment 20 of the SSAR and to review the DFSER and Reference 7. This review covered significant design changes in the SSAR (through Amendment 34) and closure of all open items in the DFSER. It also included a review of written responses by GENE to numerous questions and concerns raised by the subcommittee.

During this time our subcommittee on Improved Light Water Reactors held several meetings to complete its review of the EPRI URD. In addition, ABWR-related meetings were held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, Severe Accidents, Safeguards and Security, and our Ad Hoc Subcommittee on Design Acceptance Criteria (DAC). We did not review most of the CDM portion of the applica-

April 14, 1994

tion because we were assured by the NRC staff that it did not contain design features and requirements beyond those found in the SSAR. We did, however, review and comment (Ref. 8) on the viability of the DAC process as a suitable method for establishing future design acceptance requirements in certain areas (i.e., human factors engineering, radiation protection, piping design, and instrumentation and control). We also reviewed the CDM related to these DAC areas.

During our review of the ABWR SSAR, we considered the design-specific requirements which relate to the various evolutionary and advanced light water reactor policy, technical, and licensing issues included in SECY-90-016 (Ref. 9) and its successor, SECY-93-087 (Ref. 10). These issues incorporate staff positions that deviate from or are not embodied in current regulations. Their resolutions will become "applicable regulations" through incorporation into the design certification rule for the ABWR. We have commented previously (Refs. 11 and 12) concerning these issues.

ACRS Conclusion Concerning ABWR Safety

Based on the results of our review of those portions of the GENE ABWR application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the U.S. version of the ABWR standard design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Additional comments by ACRS Members Carlyle Michelson and Charles J. Wylie are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Carlyle Michelson and Charles J. Wylie

Although the Committee has arrived at a favorable conclusion concerning ABWR safety with which we agree, it is our view that this report should discuss the resolution of various issues that were considered by the Committee (Refs. 4 and 5) prior to reaching the favorable conclusion. Some of the resolutions were based on findings that were unanticipated and led to significant design changes. We believe that these findings should be made available

to those who must make the final safety and design certification decisions.

As an example, it was found that the rupture of an 8-inch pipe in the non-safety-grade Reactor Water Cleanup (CUW) System which is housed inside of secondary containment creates serious environmental disruption throughout the three separate divisional areas of secondary containment which house redundant portions of the Emergency Core Cooling System (ECCS). Since this 8-inch pipe contains reactor coolant at operating temperature and pressure, the break results in an immediate loss of reactor coolant until isolated and it requires an ECCS response. Steam from the break permeates the entire secondary containment because the divisional barrier doors are forced open by a buildup of steam pressure. This occurs before the primary containment isolation valves for the CUW system have time to close. A similar situation exists for the Reactor Core Isolation Cooling (RCIC) System; however, the resulting environmental conditions for most locations are bounded by those produced by the CUW 8-inch pipe break.

Since these pipe break events cannot be confined, GENE now proposes that safety-related equipment inside of the ABWR secondary containment be environmentally qualified for steam at 15 psig. and about 248°F. It is our view that this is an acceptable, although undesirable, alternative to a design which provides separation barriers and pressure relieving pathways that are capable of isolating a sufficient amount of ECCS equipment from the harsh environment. In addition, GENE has added a third break isolation valve in the 8-inch CUW supply line and located it inside of primary containment. This valve can be closed after the blowdown is over to ensure the interruption of any prolonged loss of ECCS water to secondary containment. It is needed only if both primary containment isolation valves fail to fully close due to the severe blowdown loads or other challenges common to both valves. The added environmental qualification and the third valve are new features.

References:

1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 17, 1989, from Charles L. Miller, NRC Office of Nuclear Reactor Regulation, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, August 1989
3. ACRS report dated November 24, 1989, from Forrest J. Remick, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Module I of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design

4. ACRS report dated July 18, 1991, from David A Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design
5. ACRS report dated April 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design
6. U. S. Nuclear Regulatory Commission, NUREG-1469, "Draft Final Safety Evaluation Report Related to the Design Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor," October 1992
7. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," December 1993
8. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design Approval
9. SECY-90-016, dated January 12, 1990, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
10. SECY-93-087, dated April 2, 1993, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
11. ACRS report dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements.
12. ACRS report dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, 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

May 11, 1994

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

Dear Chairman Selin:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE ASEA BROWN BOVERI -
COMBUSTION ENGINEERING APPLICATION FOR CERTIFICATION OF
THE SYSTEM 80+ STANDARD PLANT DESIGN

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we completed our review of the ASEA Brown Boveri - Combustion Engineering (ABB-CE) application for certification of the System 80+ standard plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review, we had the benefit of discussions with representatives of the NRC staff, ABB-CE and its contractors, Duke Engineering and Services, Inc., and Stone and Webster Engineering Corporation. We also had the benefit of the documents referenced.

System 80+ Application

The application for certification of the System 80+ design was filed on March 30, 1989, under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). In its letter of August 21, 1989, CE (which has been referred to as ABB-CE since May 26, 1992, as a result of CE becoming a subsidiary of ABB) stated that the application may be considered to have been submitted pursuant to 10 CFR 52.45 (Ref. 2). The application was docketed on May 1, 1991, and assigned Docket No. 52-002.

The application is based on the CE Standard Safety Analysis Report - Design Certification (CESSAR-DC), which describes the design of the facility and the site-specific interface requirements. The CESSAR-DC was originally submitted on March 30, 1989. Subsequently, ABB-CE supplemented the information in CESSAR-DC through a number of amendments. The last amendment that we received was Amendment V dated April 29, 1994. ABB-CE also submitted certified design material (CDM) (Ref. 3) on December 31, 1993, which contains Tier 1 design information which ABB-CE proposes to have certified under 10 CFR Part 52 by design certification rulemaking.

System 80+ Design Description

The ABB-CE System 80+ standard plant is designed for use at either single-unit or multiple-unit sites. In accordance with 10 CFR 52.47(b)(1), the design scope must provide an essentially complete nuclear power plant design except for site-specific elements of the design, such as the service water intake structure and the ultimate heat sink. The design evolved from the CE System 80 plant design. Three units of the System 80 design (Palo Verde Units 1, 2, and 3) have been licensed to operate in the United States.

The CESSAR-DC states that the Electric Power Research Institute (EPRI) Evolutionary Light Water Reactor Utility Requirements Document (URD) was used as a guide for the design of the System 80+ plant. Although there are some remaining differences between the System 80+ design and the EPRI URD, we do not view these differences to be significant from a nuclear safety perspective.

Four aspects of the plant design, i.e., piping design, radiation protection, instrumentation and control (I&C) design, and human factors engineering for the design of main control room and remote shutdown panel, will be completed by the Combined Operating License (COL) applicant/holder using a staff-approved design process described within the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). These ITAAC, which will be a part of the CDM, appear to be an appropriate use of the "Design Acceptance Criteria" process, which we discussed in our report of January 14, 1994 (Ref. 4).

The System 80+ nuclear steam supply system (NSSS) consists of a pressurized water reactor (PWR) with two primary coolant loops utilizing vertical U-tube steam generators. Each loop has two reactor coolant pumps. A pressurizer is connected to one of the loops. The NSSS also includes related auxiliary and engineered safety feature (ESF) systems.

The rated core thermal power is 3914 MWt. The design core thermal power, at which accidents are evaluated, is 3992 MWt. The reactor core consists of 241 16x16 Zircaloy-clad fuel assemblies and 93 control element assemblies.

The reactor containment is a 200 foot diameter spherical steel shell that is completely enclosed by a reinforced concrete Shield Building. The lower elevations of this building (the subsphere) house the four physically separated trains of shutdown cooling and ESF mechanical equipment.

The Shield Building is located within the Nuclear Island structure which also contains the fuel pool area, the maintenance outage area, the main steam valve enclosure, the two Class 1E emergency

diesel generators and their dedicated batteries, and the control complex for the plant.

The Turbine Building and the Radwaste Building are located on opposite ends of the Nuclear Island. The Turbine Building, which contains no safety-related equipment, houses the 1800 rpm turbine generator and its auxiliary systems, and major components of the condensate and feedwater systems. The turbine generator is oriented so as to reduce the likelihood of damage to safety-related equipment in the event of turbine failure. The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant.

The component cooling water heat exchangers are located within structures in the yard which surrounds the Nuclear Island, thereby eliminating the potential for flooding within the Nuclear Island due to service water pipe breaks. The combustion turbine generator (the Alternate AC power source) and its fuel supply are also located within structures in the yard. Other yard structures include the fire pump house and associated tanks, diesel fuel oil and miscellaneous water storage tanks.

Safety Enhancement Features

The ABB-CE System 80+ design includes a number of features that we believe will enhance safety relative to past PWR designs. Some of these features resulted from the use of Probabilistic Risk Assessment (PRA) methodology by ABB-CE during the System 80+ design process. The more significant features include:

- The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. Combined with improved material specifications, this reduces concern over reactor vessel integrity.
- The pressurizer and the steam generators have larger water inventories (on a volume to MWT basis) than present PWRs. This improves plant response to most transients and reduces unnecessary challenges to safety systems. In addition, the steam generators use Inconel 690 tubing, which is expected to reduce susceptibility to tube failures.
- The safety injection system (SIS) uses four half-capacity, physically separated mechanical trains that inject directly into the reactor vessel. The SIS is designed for full-flow testing during power operation. In addition to the SIS, four safety injection tanks are provided in the design. Under design basis loss of coolant accident (LOCA) conditions, these systems meet Appendix K to 10 CFR Part 50 over the spectrum of LOCA break sizes. The reactor core is expected to remain

covered with water for breaks up to a 10 inch direct vessel injection line break.

- An in-containment refueling water storage tank with external refill capability is provided as a source of borated water for both initial injection and long-term recirculation phases of the LOCA and for manually initiated cavity flooding under severe accident conditions. The tank also serves as the heat sink for the manually actuated safety depressurization system (SDS). The SDS provides the capability to rapidly depressurize the reactor coolant system, allowing the operator to initiate primary system feed and bleed during a total loss of feedwater event.
- The emergency feedwater system (EFWS) has two physically separated divisions, each consisting of an EFWS tank, a full-capacity motor-driven pump, and a full-capacity turbine-driven pump. Each EFWS division can feed both steam generators.
- The pressure boundary for the shutdown cooling system (SCS) is rated at 900 psig. This reduces concern for intersystem LOCAs. The SCS can be interconnected with the containment spray system. The pumps from either system can serve as backup to the pumps in the other system.
- The reliability of reactor coolant pump seal cooling has been improved by the inclusion of a seal cooling pump that can be powered from the combustion turbine generator under station-blackout conditions. This air-cooled pump can also provide seal cooling during loss of normal cooling water events. This pump is in addition to the charging pumps and component cooling water supplies that normally provide for reactor coolant pump seal cooling.
- Safety-related systems and trains that perform redundant functions are physically separated by appropriate barriers that provide protection against fires, floods, and similar common-cause challenges.
- The design provides for two independent offsite power connections from a main switchyard and a separate backup switchyard. The turbine generator is designed to run back and continue carrying plant auxiliary loads in the event of separation from the grid at maximum load. This feature should reduce the frequency of reactor trips following a loss of offsite power. A combustion turbine generator provides an alternate source of AC power in the event of station blackout.
- The main control complex makes use of an evolutionary design referred to as Nuplex 80+. This complex includes the main control room, the remote shutdown room, the computer room, the

technical support center, and the I&C and equipment rooms located throughout the plant. The increased use of digital control and protection systems in this design offers the potential for improving both the operator interface with the plant and the reliability of control and protection systems. The design also reduces the amount of electrical cabling, thereby reducing the potential for fire in safety-related areas.

- The 3.4 million cubic feet free volume reactor containment is large and has a higher pressure capability under severe accident conditions (estimated median ultimate containment failure pressure of 172 psia at 290°F) than most operating PWRs. These features provide added protection against early severe accident containment challenges such as hydrogen combustion and direct containment heating. They also increase the time to late containment failure due to overpressure. Provision has been made for limited unfiltered containment venting, although venting is not expected to be needed for most severe accident conditions.
- The containment design provides the capability for flooding a large (relative to current PWRs) lower reactor cavity debris spreading area prior to vessel breach. This flooding capability can be activated independently of AC power sources. In addition, a thick basemat made with ablation resistant concrete is used.
- The design provides a massive reactor cavity/reactor vessel support structure. This structure is intended to withstand the pressure that could result from direct containment heating or ex-vessel fuel coolant interaction. A convoluted de-entrainment pathway is provided between the cavity and the upper containment to minimize the expulsion of corium out of the cavity during a core melt ejection event.
- The design includes a hydrogen mitigating system employing manually activated glow plug igniters at 40 locations (two independently powered igniters per location) in the containment. Care was used in the design to vent those compartments where hydrogen could accumulate.
- The containment spray system (CSS) uses two independent trains. A connection is provided to the CSS for an emergency containment spray backup system, consisting of a cooling pond water source, and a portable pump capable of being driven independently of AC power sources.
- Design features that minimize shutdown and low power operation risk were analyzed with the result that no significant design

vulnerabilities were found for accidents involving shutdown and low power operations.

Chronology of ACRS Review

Our review of the System 80+ application commenced after it was filed in March 1989. We held a series of Subcommittee meetings between April 1990 and February 1993. The staff issued a Draft Safety Evaluation Report (DSER) on October 1, 1992 (Ref. 5). In December 1993, the ACRS Subcommittee on ABB-CE Standard Plant Designs began a series of meetings dedicated to the final review of the CESSAR-DC and related material. This series of meetings built upon and continued the previous ACRS activities, and provided the basis for this report. The staff issued a Final Safety Evaluation Report (FSER) on March 3, 1994 (Ref. 6). Our activities related to System 80+ are described in the attachment.

ACRS Conclusion Concerning System 80+ Safety

Based on the results of our review of those portions of the ABB-CE System 80+ application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the System 80+ standard plant design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Sincerely,

T. S. Kress
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 21, 1989, from A.E. Scherer, CE, to T.E. Murley, NRC, Subject: Design Certification of the System 80+™ Standard Design
3. Letter dated December 31, 1993, from C.B. Brinkman, ABB-CE, to USNRC Document Control Desk, Subject: System 80+™ ITAAC Submittal
4. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design

5. Letter dated October 1, 1992, from R.C. Pierson, NRC, to C.B. Brinkman, ABB-CE, Subject: Draft Safety Evaluation Report (DSER) of Nuclear Regulatory Commission (NRC) Staff Review of Combustion Engineering (ABB-CE) Standard Safety Analysis Report for Design Certification of System 80+ (NUREG-1462)
6. Letter dated March 3, 1994, from James M. Taylor, NRC Executive Director for Operations, to the NRC Commissioners, Subject: Advance Copy of the Final Safety Evaluation Report (FSER) on the ABB-Combustion Engineering System 80+ Standard Design Certification and Certified Design Material (CDM)

Attachment:

Chronology of ACRS Review

ATTACHMENT - CHRONOLOGY OF ACRS REVIEW

Discussions during the following ACRS Subcommittee and Full Committee meetings included the listed topics on ABB-CE System 80+:

April 3, 1990 - Advanced PWR Subcommittee

Licensing Review Basis (LRB) document, reactor coolant system, engineered safety feature systems, containment, Nuplex 80+, and probabilistic risk assessment (PRA)

September 21, 1990 - Advanced PWR Subcommittee

Use of operational experience at existing Combustion Engineering plants, including reactor coolant pump impellers, resistance temperature detectors, heated junction thermocouples, upper guide structure, safety injection nozzle thermal sleeves, steam generator geometry and operating parameters, fire protection, security, and flood design

November 1, 1990 - Advanced PWR Subcommittee

Licensing Review Basis Document. An ACRS report was issued on November 14, 1990, regarding the LRB document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor.

February 6, 1991 - Joint meeting of the Subcommittees on Computers in Nuclear Power Plant Operations, and Instrumentation and Control (I&C) Systems on computer applications in advanced plant designs

Nuplex 80+ software reliability

March 6, 1991 - Advanced PWR Subcommittee

Design basis accident analysis, and seismic methodologies

September 4, 1991 - Advanced PWR Subcommittee

Piping layout, Nuplex 80+ advanced control room design, and PRA

December 3 and 4, 1991 - Joint meeting of the Subcommittees on Advanced PWR and Computers in Nuclear Power Plant Operations with Westinghouse and CE regarding digital computer experiences at nuclear power plants

Core Protection Calculator improvements and remote multiplexing

March 4, 1992 - Joint meeting of the Subcommittees on Computers in Nuclear Power Plant Operations, I&C Systems, and Human Factors with representatives of EPRI, CE, Westinghouse, and Software Engineering Institute

Nuplex 80+ control room design bases and features

September 10-12, 1992 - 389th ACRS meeting

Defense against common-mode failures in digital I&C systems

February 10, 1993 - Advanced PWR Subcommittee

Design overview, human factors engineering, protection for common-mode software failure of I&C systems, physically based radiological source term, and radiological equipment qualification

December 8, 1993 - ABB-CE Standard Plant Designs Subcommittee

Combustion Engineering Standard Safety Analysis Report-Design Certification (CESSAR-DC) and NRC staff Final Safety Evaluation Report (FSER) Chapters 7, 8, and 18

February 9, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 4, 10, 11, 12, 13, 14 (section 2), and 17

March 8 and 9, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 2, 3, 14 (section 3), and 19

March 17, 1994 - Palo Verde Nuclear Generating Station Site Visit

Several members of the ACRS attended a fact-finding visit which included familiarization with the plant, site arrangement, and operating history of the System 80 design

April 5 and 6, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 1, 5, 6, 9, 15, 16, and CESSAR-DC Appendix A (FSER Chapter 20). In addition, during this meeting the Subcommittee reviewed the applicant's evaluation that, for the worst credible accident, the dose at the site boundary (one-half mile from the reactor) will remain below the Environmental Protection Agency's lower Protective Action Guideline of 1 rem. This is expected to be the subject of a separate Committee report.

May 5-7, 1994 - 409th ACRS Meeting

ABB-CE and NRC staff responses to questions asked by ACRS members during previous Subcommittee meetings

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

November 10, 1994

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

Dear Mr. Taylor:

SUBJECT: NRC TEST AND ANALYSIS PROGRAMS IN SUPPORT OF AP600 AND
SBWR ADVANCED LIGHT WATER REACTOR PASSIVE PLANT DESIGN
CERTIFICATION REVIEWS

During the 414th and 415th meetings of the Advisory Committee on Reactor Safeguards, October 6-7 and November 3-4, 1994, we discussed the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification reviews for the Westinghouse AP600 and GE Nuclear Energy (GENE) Simplified Boiling Water Reactor (SBWR) advanced light water reactors. During these meetings, we had the benefit of discussions with representatives of RES. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on August 25-26, 1994, to discuss this matter. We also had the benefit of the documents referenced.

In the absence of a full-scale test facility, an understanding of the thermal hydraulic behavior of a passive plant design will depend on the use of computer codes. The NRC staff has decided to modify RELAP5/MOD3 for its confirmatory thermal hydraulic analysis of the AP600 and SBWR designs. The important phenomena the code must simulate should be delineated in the Phenomena Identification and Ranking Table (PIRT), thus allowing one to formulate integral and separate effects experiments that will yield appropriate data for code validation. Code validation should be an integrated process involving code development, experimentation, and an understanding of the physics of two-phase flow and heat transfer.

The major objective of the thermal hydraulic code development effort should be to produce a code capable of predicting the behavior of a full-scale nuclear power plant with acceptable uncertainties. For existing nuclear plant designs, we have had the benefit of many integral and separate effects experiments at a wide variety of scales to help arrive at an estimate of the uncertainties in the code predictions. We are now dealing with two passive plant designs which evidence more complex thermal hydraulic system

dynamics, and for which there is a paucity of relevant experimental data. There are several causes for this more complex dynamic behavior: (1) steam condensation at low pressure, (2) use of gravity-driven coolant injection, and (3) the existence of many components and complex hydraulic paths that give the system many degrees of freedom. Understanding this dynamic behavior requires evaluation of scale distortion effects and dynamic characteristics in the various test facilities. In this regard, two questions should be addressed and resolved: (1) is the evolution of a particular transient influenced by configurational and/or scale distortions, and (2) do configurational and/or scale distortions in the various test facilities preclude simulation of some important dynamic effects while introducing other dynamic effects that may not be important in a full-scale plant design? To address these questions, a top-down scaling analysis must be performed.

The NRC staff has test and analysis programs under way to address issues arising during its evaluation of the AP600 and the SBWR designs. The AP600 evaluation will be supported by testing at the Japan Atomic Energy Research Institute ROSA-V facility and the use of RELAP5/MOD3. The SBWR evaluation will be supported by testing at the Purdue University PUMA facility and the use of RELAP5/MOD3. We believe that the use of RELAP5/MOD3 for both AP600 and SBWR simulations will lead to the development of a more robust computational tool. Both programs are discussed below and some comments about the technical direction of these programs are provided.

AP600 Program

The PIRT in support of the AP600 analysis has not yet been completed. There is no indication that a PIRT was utilized for allocating resources, for assigning test objectives, or for developing the test matrices. It is necessary to complete the PIRT and confirm it on the basis of relevant scaling groups. To ensure that RELAP5/MOD3 can simulate the high ranking phenomena, specific tests in the test matrix should be associated with the high ranking phenomena in the PIRT. By doing this, all important phenomena will be addressed.

The PIRT and a proper scaling analysis for the AP600 would cover all test facilities for AP600. Unfortunately, the scaling efforts conducted for the OSU, SPES, and ROSA-V test facilities were not coordinated. The global scaling of the AP600 design, including consideration of the dynamic interactions between the major system components (pressure vessel, core makeup tank, pressurizer, steam generators, passive residual heat removal system, and accumulators), was omitted. Depressurization is not scaled, even though the methods for doing so are known. The scaling analysis for OSU, while still incomplete, could serve as a model for ROSA and SPES.

Direct counterpart tests in ROSA, OSU, and SPES are not possible. This makes it difficult to extrapolate the observed thermal hydraulic behavior to full scale. A well-planned effort to integrate experiments with code improvement and assessment is needed to quantify uncertainties. At present, RELAP5/MOD3 predicts strong oscillations both when they are observed in tests and when they are not. Consequently, the calculated behavior can neither be attributed conclusively to numerical nor physical effects. The mechanisms by which the various observed modes of oscillation are initiated and maintained need to be understood so that their potential influence on the thermal hydraulic behavior of the AP600 can be evaluated. The judicious selection of test conditions for the facilities, together with the conduct of a careful data analysis and scaling, should provide a satisfactory solution.

The demonstrated propensity for condensation oscillation events in the AP600 points to a need to identify both the likelihood and damage potential of water hammer events. Furthermore, the influence of thermal stratification on the thermal hydraulic behavior of the AP600 also remains to be evaluated.

SBWR Program

The objective of the PUMA test program is to obtain data for assessing computer code simulation of important SBWR-specific phenomena. The focus of this test program is on the operability of the passive cooling systems and their interactions with the reactor vessel.

Again, a PIRT has not been completed. The PIRT effort should be brought to a close so that a proper evaluation of PUMA and the GENE test facilities (GIST, GIRAFFE, and PANDA) can be made.

Scaling of phenomena identified in the Purdue University preliminary PIRT has been a major part of the PUMA test program. At present, the scaling effort has primarily focused on the details of local phenomena whereas global scaling appears to be incomplete. To preclude atypicalities in the interactions of the various systems and to help determine an appropriate set of initial and operating conditions for the PUMA system, the scaling of the global dynamic component interactions (among the reactor vessel, drywell, wetwell, PCCS, ICS, and GDCS) should be completed before the facility design is frozen.

We are pleased to see that one of the PUMA program principal investigators is a code developer. Input from a code developer on the selection of instrument type, number, and location will yield a much more useful set of data for code assessment.

The PUMA facility will allow testing that both overlaps and extends the accident period covered by the GENE test facilities (GIST, GIRAFFE, and PANDA), while allowing the simulation of a broad spectrum of postulated accidents. This should be helpful in confirming the validity of the results obtained at the GENE facilities.

The following comments are specific to the PUMA program:

- The current plan is to measure the heat transfer characteristics and infer the noncondensable gas concentration. We would like to point out that knowledge of the noncondensable gas distribution is fundamental and necessary if one is to avoid compensating errors in the computational process. We recommend that the noncondensable gas concentration be measured directly at several locations.
- The test matrix does not include a long-duration test. We believe it should because the SBWR containment performance requirement is 72 hours, which scales to 144 hours of PUMA test time.
- Since the interface temperature of the suppression pool is directly coupled to the containment pressure, an evaluation of thermal stratification in the pool is needed.
- Some tests should be conducted with initial nitrogen concentrations in the drywell to evaluate the impact of steam line breaks outside containment.
- The planning of the PUMA experiments should include consideration of phenomena arising as a consequence of failures of active mitigating systems.
- Data analysis and evaluation are not part of the contract with Purdue University. This is unfortunate because in this case the principal investigators at Purdue University are highly qualified for such a task. Further, those conducting the testing can bring valuable insights to the process. We recommend that the contract with Purdue University be modified to include a data analysis and evaluation task.

Technical Oversight

The RES staff now plans technical oversight of thermal hydraulic research for the AP600 and the SBWR through the Advanced Light Water Reactor Thermal Hydraulic Research Integration Group (ATRIG). This unwieldy ATRIG is not the technical oversight recommended by the ACRS in the past and subsequently approved by the Commission. Lessons learned from the CSAU program should be remembered. A small (5 or 6 members) cohesive group with well-qualified leader-

ship is needed to integrate the technical issues of scaling, data collection, data analysis, and code development.

Sincerely,

T. S. Kress
Chairman

References:

1. "Summary of the LSTF Characterization Tests Performed in Conjunction with the ROSA/AP600 Experiments," R. A. Shaw, et al., Draft report dated August 1, 1994, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
2. U. S. Nuclear Regulatory Commission, Draft NUREG/CR, PU-NE 94/1, Subject: Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR, July 1994, transmitted by memorandum dated August 4, 1994, from J. T. Han, Office of Nuclear Regulatory Research
3. Memorandum dated August 8, 1994, from M. Ishii, Purdue University, to J. Han, U. S. Nuclear Regulatory Commission, transmitting replacement pages for the report, "Preliminary Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR"
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6066, EGG-2705, "Scaling and Design of LSTF Modifications for AP600 Testing," T. J. Boucher, et al., August 1994
5. SECY-94-138, memorandum dated May 20, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Confirmatory High Pressure Integral System Testing of the Westinghouse AP600 Safety Systems
6. "Quick Look Report for ROSA/AP600 Experiment AP-CL-03," R. A. Shaw, et al., undated rough draft, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
7. Advisory Committee on Reactor Safeguards Report, dated November 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification

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

October 14, 1994

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

Dear Chairman Selin:

SUBJECT: POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA
GENERATED DEBRIS

During the 414th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 1994, the Committee was briefed by the NRC staff on the emergency core cooling system (ECCS) recirculation strainer blockage issue raised by the event that occurred at the Barsebäck plant in Sweden on July 28, 1992. We heard previous briefings in January 1993, July 1993, and April 1994. During the present meeting, the staff discussed (1) a proposed Revision 2 to Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," (2) the contractor draft report NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," which has been issued for public comment, and (3) the staff plan for issuing a generic letter on this matter in August 1995. A representative of the Boiling Water Reactor Owners Group (BWROG) presented industry views and actions. We also had the benefit of the documents referenced.

The Barsebäck event involved BWR ECCS strainer blockage caused, in this case, by debris dislodged as a result of inadvertent safety valve discharge into the drywell. Our assessment of this event indicates that strainer blockage due to accident generated debris is an important safety issue for at least some BWRs and that strainer blockage was not adequately addressed in the 1985 resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The present version of RG 1.82, which formed the basis for resolution of USI A-43, deals principally with PWR ECCS sumps and provides prescriptive detailed information for PWR designs acceptable to the staff (design sketches, dimensions, etc.). The staff apparently plans to provide similarly prescriptive design information for BWR suppression pool ECCS suction strainers through its planned revision to RG 1.82.

Both the staff and BWROG agree that this is a compliance issue. However, BWR licensees may be reluctant to make plant modifications

(beyond those interim compensatory measures required by NRC Bulletin 93-02 and its supplement) until the staff completes its deliberations on the revision to RG 1.82. Some obvious actions that licensees could have taken after the Barsebäck event to protect against the effects of LOCA generated debris are: (1) replacement of fibrous insulation with reflective metallic insulation, (2) installation of strainers with larger screen areas or other improvements, (3) installation of differential pressure sensors on ECCS pump suction strainers to detect strainer blockage, and (4) installation of strainer cleaning systems. It is our understanding that most European operators of BWRs have made or are making some or all of these modifications.

We question whether the approach the staff is taking will result in timely corrective actions. It seems to us that the onus should have been on the BWR licensees to evaluate the vulnerability of their plants to ECCS strainer blockage due to LOCA generated debris and to propose appropriate plant-specific modifications to deal with the issue. The survey performed by the BWROG in 1992 indicated that each plant is unique with respect to the nature of and potential for debris generation and strainer design and backflush capability. Therefore, plant-specific solutions are needed.

Draft NUREG/CR-6224, which was not initiated until September 1993, provides valuable insights and confirms quantitatively much that was qualitatively known and understood shortly after the Barsebäck event. A troubling insight among these is the indication that ECCS strainer blockage contributes significantly to core damage frequency (CDF) for the reference plant and similar BWRs. However, the authors of the report point out that there are many limitations and uncertainties associated both with the analysis that led to the reference plant results and with extrapolating these results to other BWRs.

Three comments evolved from our review. First, we are concerned by the implications of the prediction that the contribution due solely to strainer blockage is over three times the CDF represented in the reference plant Individual Plant Examination (IPE). We encourage the staff to examine the treatment of LOCA generated debris in other plant IPEs.

Second, we believe that the scope of draft NUREG/CR-6224 should be expanded to look at debris generation resulting from the flow of steam/water mixtures at some distance from the LOCA break location. This flow and pressure may dislodge pipe insulation, particularly if pressure equilibration is slow across the insulation, and may damage other debris producing targets such as the very large containment air handling units in the drywell.

Third, there is the potential for damaging ECCS pump seals or causing a loss of bearing cooling due to LOCA-generated fibrous and/or particulate matter. It is our understanding that most or all operating BWRs use pump discharge water for seal injection and bearing cooling. This issue, which we first raised in our letter of September 16, 1985, to the NRC Executive Director for Operations (EDO), has been discussed with the staff during our recent series of meetings. We believe that this issue needs to be evaluated and resolved as a part of the resolution of the ECCS strainer blockage issue.

In summary, we are concerned by the slow pace at which this important safety issue is being addressed. We recommend that the EDO and his senior staff critically review the current action plan and take the necessary steps to facilitate prompt resolution.

We plan to continue to monitor the NRC staff and industry's resolution of this issue.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated August 26, 1994, from Joseph A. Murphy, Office of Nuclear Regulatory Research, NRC, to Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, Subject: Review of DG-1038, Proposed Revision 2 to RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following A Loss-of-Coolant Accident"
2. U. S. Nuclear Regulatory Commission, NUREG/CR-6224, Draft Report for Comment, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," August 4, 1994
3. U. S. Nuclear Regulatory Commission, OMB No. 3150-0011, NRCB 93-02: Debris Plugging of Emergency Core Cooling Suction Strainers, May 11, 1993
4. Letter dated September 16, 1985, from David A. Ward, Chairman, ACRS, to William J. Dircks, Executive Director for Operations, NRC, Subject: ACRS Review of Proposed Resolution for USI A-43, "Containment Emergency Sump Performance" and Regulatory Guide 1.82, Revision 1, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident"

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

January 21, 1994

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

Dear Mr. Taylor:

SUBJECT: DRAFT FINAL RULEMAKING PACKAGE DEALING WITH EMERGENCY
PLANNING REGULATIONS

During the 405th meeting of the Advisory Committee on Reactor Safeguards, January 6-7, 1994, we reviewed the subject rulemaking package. We had the benefit of discussions with representatives of the NRC staff and the referenced document.

The proposed final rule would amend the existing Appendix E to 10 CFR Part 50, as follows:

- Reduce the frequency of required participation of States in the ingestion pathway portion of emergency planning exercises from at least once every five years to at least once every six years.
- Delete the requirement that all States within the emergency planning zone for a given site fully participate in an off-site exercise for that site at least once every seven years.
- Clarify and update the language on participation by State or local governments in the biennial off-site exercise.

The language in the proposed final rule reflects the staff resolution of the few public comments it received. One of the benefits of the proposed rule would be a greater compatibility between NRC and the Federal Emergency Management Agency requirements.

We believe that the proposed rule will reduce the regulatory burden somewhat with no significant effect on the public health and

Mr. James M. Taylor

2

January 21, 1994

safety. Accordingly, we recommend that the proposed final rule be issued.

Sincerely,

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", with a large, stylized initial "J" and a long, sweeping underline.

J. Ernest Wilkins, Jr.
Chairman

Reference:

Memorandum dated December 1, 1993, from W. Minners, Office of Nuclear Regulatory Research, for J. T. Larkins, ACRS, Subject: ACRS Review of Draft Final Rulemaking Package Clarifying the Emergency Planning Regulations Relating to Exercises, with enclosures

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

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 projected 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."

"... 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

July 13, 1994

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

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

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

June 14, 1994

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

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 410th meeting of the Advisory Committee on Reactor Safeguards, June 9-10, 1994, we discussed the proposed staff approach for resolving Thermo-Lag fire barrier issues with representatives of the NRC staff and Nuclear Energy Institute (NEI). Our Subcommittee on Auxiliary and Secondary Systems reviewed this matter during a meeting on June 8, 1994. We also had the benefit of the documents referenced. This report is in response to the March 18, 1994 Staff Requirements Memorandum.

We agree with the staff's view that an immediate order to require upgrading of inadequate Thermo-Lag fire barriers is not needed based on defense-in-depth arguments and the fact that compensatory measures are already in place at those plants that have not resolved their Thermo-Lag problems.

In SECY-94-127, the staff describes the following four options for resolving the Thermo-Lag fire barrier issues:

Option 1 - Require Compliance with Existing NRC Fire Barrier Requirements

Option 2 - Develop Guidance for Rating Fire Barriers Based Upon a Range of Combustible Loadings for Fire Endurance Tests

Option 3 - Develop a Performance-Based Approach Using a Lead Plant

Option 4 - Develop a Performance-Based Fire Protection Rule

We support the staff recommendation described as Option 1, which includes provisions for plant-specific exemptions as permitted in the current regulations. However, we believe that exemptions under Option 1 should not be limited to those permitted by precedent.

Fire-analysis techniques have advanced substantially since the current fire protection regulations were promulgated. These advances justify a reexamination of the bases for granting exemptions. We recommend that, in the near term, the staff and industry work toward the development of generic guidelines for using performance-based approaches to justify exemptions.

We are advocates of risk-based regulation and therefore support the staff's plan, described in SECY-94-090, to develop risk-based and performance-oriented fire protection regulations and recommend that any such regulatory framework include consideration of fire risk during shutdown conditions.

Additional comments by ACRS Member Ivan Catton are presented below.

Sincerely,

T. S. Kress
Chairman

Additional Comments of ACRS Member Ivan Catton

While I agree with some of what is said in the above report, I do not understand why the implementation of Option 2 is considered to be so complex. The computational tools are available to support the selection of Option 2 as a means to resolve the Thermo-Lag issues without resorting to a large number of exemptions. There are examples of how this can be done. Further, most of what must be done will support the effort to achieve a performance-based fire protection regulation. I believe it is time to follow the lead of other countries (e.g., Sweden, Australia, and others) in moving toward realistic performance-based fire protection regulation.

References:

1. SECY-94-127 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Options for Resolving the Thermo-Lag Fire Barrier Issues
2. SECY-94-128 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Status of Thermo-Lag Fire Barriers
3. 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

4. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
5. SECY-94-024 dated February 4, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Resolution of Issues Concerning Thermo-Lag Fire Barriers
6. SECY-93-143 dated May 21, 1993, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: NRC Staff Actions to Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program
7. Memorandum dated March 25, 1994, to Holders of Operating Licenses from Luis A. Reyes, Office of Nuclear Reactor Regulation, NRC, Subject: Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area (Supplement 1 to Generic Letter 86-10, "Implementation of Fire Protection Requirements")
8. Letter dated March 4, 1994, from Alex Marion, Nuclear Management and Resources Council, to C. McCracken, Office of Nuclear Reactor Regulation, NRC, transmitting NUMARC Industry Application Guide to Evaluate Thermo-Lag Fire Barriers (Draft D)

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

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

References:

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

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

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

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. 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

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

May 11, 1994

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

Dear Mr. Taylor:

SUBJECT: DRAFT POLICY STATEMENT ON THE USE OF PROBABILISTIC RISK
ASSESSMENT METHODS IN REACTOR REGULATORY ACTIVITIES

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we reviewed the current draft Policy Statement on agency usage of probabilistic risk assessment (PRA). We had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We are in general agreement with the Policy Statement. It appears to present an appropriate position on the use of PRA in the regulatory process. We are, however, concerned with some aspects of the Policy.

Some provisions of the Policy Statement are crafted in rather weak language. For example, we believe that in Item (2) of Section II, Policy Statement, the word "may" ought to be replaced by "should" to make a commitment to increase the use of PRA to help eliminate unnecessary conservatism associated with current regulatory requirements.

The Policy is very general and does not provide any specific guidance or plan for the expanded use of PRA in regulatory activities. This has apparently been relegated to an "implementation plan" which is referred to in the Policy Statement. We hope that this plan will provide some specific and definitive elements to guide the use of PRA in the regulatory process. We recommend that the implementation plan be submitted for public comment along with the Policy Statement.

The draft Policy Statement seems to draw a distinction between the traditional regulatory process (commonly known as "deterministic") and the PRA approach. This common perception causes some in the regulatory arena to be skeptical of and reluctant to embrace the PRA approach. However, we believe that treating the PRA approach as a distinct and unique method compared to the traditional approach is inappropriate and misleading. We believe that the PRA approach should be considered as an extension and enhancement of traditional regulation rather than a separate and different technology. Certainly, the deterministic approach is replete with implied elements of probability, from the selection of accidents to

be analyzed (e.g., reactor vessel rupture is too improbable to be considered) to the requirements for emergency core cooling (e.g., safety train redundancy and protection against single failure). The PRA approach enhances traditional approaches by considering risk in a coherent and complete manner, thereby providing a method to quantify the overall level of safety.

We agree that there are uncertainties, limitations, and omissions with the PRA approach. However, we think it is important to understand that these uncertainties are derived from knowledge limitations. These knowledge limitations were not created by PRA, but rather were exposed by it. These limitations existed during the traditional regulatory approach, some were unknown, others only vaguely understood. Attempts were made to accommodate these limitations by imposing prescriptive and what was hoped to be conservative regulatory requirements. The PRA approach has exposed these limitations and has provided a framework to assess their significance and assist in developing a strategy to accommodate them in the regulatory process. We are pleased that these issues are identified in the Policy Statement and that they are being addressed in the implementation plan.

One of the more important shortcomings of PRA use was not identified in the Policy Statement. This is the misuse and misapplication of PRA results stemming from an incomplete and/or flawed analysis. While those in the nuclear regulatory arena have done an excellent job in many instances in applying and using PRA, there have been examples where this has not been the case. Among the more important of these are some of the cost/benefit analyses for backfits. We recognize that these analyses are difficult. We urge the staff to assign high priority in the implementation plan to improving and adding consistency to cost/benefit analyses.

We further believe that the implementation plan needs to address the need for PRA research to help assure that the PRA state-of-the-art is at a level consistent with the intended PRA usage in the agency. We intend to further consider the area of PRA research needs in the near future.

In conclusion, we reiterate our support for the overall thrust of the PRA Policy Statement and the allocation of resources to implement it. We would like to be kept informed of the progress in developing the implementation plan.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum (Undated) from James M. Taylor, Executive Director for Operations, for The Commissioners, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities, received May 5, 1994 (Predecisional)
2. Memorandum dated April 14, 1994, from Martin J. Virgilio, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: PRA Draft Policy Statement, with Predecisional Enclosure
3. U.S. Nuclear Regulatory Commission, Policy Statement dated January 18, 1979, Subject: NRC Statement on Risk Assessment and The Reactor Safety Study Report (WASH-1400) In Light of the Risk Assessment Review Group Report

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

December 20, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL DRAFT REGULATORY GUIDE, DG-1023,
"EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY
UPPER-SHELF ENERGY LESS THAN 50 FT-LB"

During the 416th meeting of the Advisory Committee on Reactor Safeguards, December 8-10, 1994, we discussed the subject regulatory guide with representatives of the NRC staff. We provided comments on an earlier version of this regulatory guide in a letter dated July 15, 1993. We also had the benefit of the documents referenced.

The data from ongoing reactor surveillance programs suggest that the Charpy upper-shelf energy will decrease to less than the present regulatory limit of 50 ft-lb for a number of operating reactor pressure vessels. The need for a guide for the evaluation of the integrity of such pressure vessels was highlighted during the discussion of the Yankee Rowe reactor pressure vessel. We believe that this guide will prove useful to the licensees and the staff, and endorse its adoption.

Additional comments by ACRS members Ivan Catton, Thomas S. Kress, Dana A. Powers, and Robert L. Seale are presented below.

Sincerely,

A handwritten signature in cursive script, reading "T. S. Kress", is positioned above the printed name.

T. S. Kress
Chairman

Additional Comments by ACRS Members Ivan Catton, Thomas S. Kress, Dana A. Powers, and Robert L. Seale

The final draft regulatory guide proposes an analysis to demonstrate the safety of reactor pressure vessels even though irradiation has reduced fracture toughness of the vessel material below a Charpy upper-shelf energy of 50 ft-lb. There has, however, not been an attempt to quantify the safety margins produced by the compound safety factors on loads, initial flaw size, and material properties involved in the analysis. Such quantification of margins may be essential in defending the assertion that the revised analysis leads to equivalent safety yet does not impose undue burden on licensees. A quantitative understanding of margins may also be of use in assessing the importance of continuing degradation of material properties with further irradiation. Though publication of the final draft regulatory guide ought not be delayed, an effort, building upon results of research, should be undertaken to develop a quantitative understanding of safety margins indicated by the revised analysis.

References:

1. Letter dated November 3, 1994, from L. Shao, Office of Nuclear Regulatory Research, NRC, to J. Larkins, Executive Director, ACRS, Subject: Request For ACRS Review of Draft Regulatory Guide, DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb"
2. Letter dated July 15, 1993, from J. E. Wilkins, Jr., Chairman, ACRS, to J. Taylor, Executive Director for Operations, NRC, Subject: Proposed Draft Regulatory Guides, DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb," and DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

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

February 17, 1994

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

Dear Chairman Selin:

SUBJECT: DIVERSITY

On occasion the NRC staff has proposed, and in some cases either formally or informally mandated, the use of some form of diversity as a protection against conjectured common-cause failures in redundant components or systems. In some cases this requirement has been deemed met by the use of different manufacturers of identical systems, in others functional diversity has been required, and in still others differences in design, personnel or physical principles have been required. In some of our memories the issue dates back to the subject of Anticipated Transients Without Scram, while others' memories are longer. Our most recent encounters have been in the contexts of digital controls and of water-level indicators for boiling-water reactors. The staff argument has been that hypothetical and otherwise unidentified common-cause failures are more likely to afflict identical than dissimilar systems, so that diversity per se is almost axiomatically a safety benefit. Though the argument has an incontestably solid core, we don't recall any case in which it has been quantified beyond that point.

The Commission has long been challenged by the problem of determining the proper level of safety to be reasonably required of the nuclear power industry, and the 1986 Safety Goal Policy Statement made explicit what had long been recognized—that a policy that demands any improvement in safety simply because it is an improvement is both unwise and unsupportable. The Safety Goals made explicit statements about the Commission's intent, and the subsequent crystallization of the Commission's Backfit Rule dealt with the guidelines for safety improvements beyond the "adequate safety" criterion. Continuing the same trend, the Commission is now engaged in a long-term metamorphosis into a risk-based regulatory agency, meaning that regulatory decisions will be, in the end, justified in terms of their expected impact on the health and safety of the public and the workers. Implementation of this grand strategy requires both the will and the capability to analyze regulatory proposals for their consistency with the policy, which

in turn requires increased reliance on analytical methods of risk assessment. It is precisely at that point that the diversity arguments always seem to fall short.

Of course we do not argue that diversity is always bad—only that a diversity requirement imposed by the NRC demands more justification than a flat assertion that diversity is desirable in the abstract. In any specific case the detailed arguments may force the conclusion either way, but the outcome cannot be known in advance, without analysis. The argument that analysis is not needed because there may be unspecified common-cause accidents for which diversity might be beneficial is inconsistent with the Commission's policies mentioned above.

Now we turn to a more-or-less random list of circumstances in which diversity has a negative safety impact, also in the abstract, just to provide a counterpoint to the assertion that it is always good. As we have said, the essence of rational regulation in the interest of public safety requires that, in each case, the advantages and disadvantages of a diversity requirement be weighed against each other, and the winner judged against the higher standards of either "adequacy" or the cost-benefit criteria of the Backfit Rule, as appropriate.

It is almost never true that a requirement for diversity will result in diverse instruments, components, or systems that are equally reliable. Since it would make little sense to choose the inferior of two options for the primary system, diversity will necessarily require a known and intended sacrifice in component reliability, in return for protection against hypothesized common-cause failures. Should an elevator be held up by a steel cable and a backup Manila rope? Should an airplane have a piston engine to back up its jet engines? Those are farcical cases, but the question of whether a steam-driven pump should back up an electrically driven pump addresses a known potential common-cause failure, and may yield a different answer.

Diversity increases risk by increasing complexity—simplicity is usually a safety advantage. This effect shows up through the availability, stocking, and interchangeability of parts, proliferation of operational and maintenance manuals, training of operational and maintenance personnel, interpretation of symptoms in an upset condition, a larger variety of failure modes and effects, unfamiliarity with the characteristics of the backup system if it is normally held in reserve, and so forth. There have been many industrial accidents in which complexity has introduced confounding factors that either caused or exacerbated the accident.

Diversity introduces new accident paths, and introduces components whose reliabilities are likely to be less well known than that of the primary component. Accident analysis, and therefore plant

design, becomes more difficult as one has to deal with a greater variety of potential upset paths.

When dealing with diverse or redundant sensing systems, one has always to resolve the issues of voting logic. There can be few more perilous conditions than to have comparably credible instruments that disagree. The questions of voting logic for diverse combinations of instruments are far more subtle and deep than the simple considerations that go into the traditional nuclear voting logic. Many accidents in other industries have been compounded by inappropriate voting logic.

We offer this list simply to counter the staff's apparent mindset (evident in many places) that diversity is always a desirable system attribute. As we move in fits and starts, but inevitably, toward some form of risk-based regulation, it is incumbent on the staff to make a balanced case for any diversity requirement it seeks to mandate, on the merits. We wish only to supply some of the cons that must be balanced against the pros, so the outcome is not decided by slogan.

We seek no action through this letter, only increased sensitivity of both the Commission and the staff to the fact that it is all too easy to oversimplify the case for diversity.

Sincerely,

A handwritten signature in dark ink, reading "J. Ernest Wilkins, Jr." with a stylized flourish at the end.

J. Ernest Wilkins, Jr.
Chairman

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

March 15, 1994

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

Dear Chairman Selin:

SUBJECT: NEED FOR REVIEW OF RATIONALE FOR REGULATION

During our review of source-term issues at the 407th meeting of the Advisory Committee on Reactor Safeguards, March 10-12, 1994, we were struck once again by the need to review the traditional bases for plant licensing and regulation. The source-term issues are part of the general question of the proper role for the standard design basis accidents, which is part of the issue of the often-mentioned bottom-up review of the General Design Criteria. These basic reviews of the continuing rationale for regulation never seem to assume a high priority within NRC, relative to short-term matters. We think they should.

Sincerely,

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", is written over the typed name.

J. Ernest Wilkins, Jr.
Chairman

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

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. Turbine Inspection Requirements - 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. Technical Specification Requirements for Onsite Power Sources - 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. Reactor Water Cleanup System Safety - 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. Filters or Water Separators for the Hardened Vents Installed on Operating BWR Containments - 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. Fuel-Coolant Interactions - 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. Adequacy and Use of PRA - 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

References:

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

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

September 14, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED GENERIC LETTER ON THE USE OF NUMARC/EPRI REPORT
TR-102348, "GUIDELINE ON LICENSING DIGITAL UPGRADES"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we reviewed the subject proposed generic letter. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

The proposed generic letter endorses, with two clarifications, Nuclear Management and Resources Council/Electric Power Research Institute (NUMARC/EPRI) Report TR-102348 as useful guidance for effectively implementing digital upgrades and for determining when these can be performed without prior NRC staff approval under the requirements of 10 CFR 50.59.

We basically concur with the proposed generic letter and have no objection to issuing it for public comment. However, we believe that additional clarification should be provided regarding equipment environmental compatibility. Specifically, it should be made clear in the generic letter that the environmental requirements as defined in Subsection 5.3, "Compatibility With the Environment," of the NUMARC/EPRI report include all environmental conditions resulting from internal and external events to which the equipment may be subjected. This subsection currently focuses on the need to address electromagnetic interference. We believe that any guideline which purports to cover environmental compatibility issues for replacement equipment must require that other environmental stressors such as temperature, humidity, radiation, vibration/seismic, and smoke be addressed. We note that the need to prioritize these and to verify the appropriateness of current research programs was identified in our letter of November 12,

Mr. James M. Taylor

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1992, and that you agreed. We anticipate a briefing on the results of this effort.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated August 30, 1994, from E. Doolittle, NRC Office of Nuclear Reactor Regulation, to J. Larkins, ACRS Executive Director, forwarding Proposed NRC Generic Letter on the Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades"
2. Letter dated December 22, 1993, from W. Rasin, NUMARC, to W. Russell, NRC Office of Nuclear Reactor Regulation, forwarding EPRI Report TR-102348
3. ACRS letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems
4. Letter dated December 10, 1992, from James M. Taylor, NRC Executive Director for Operations, to Paul Shewmon, ACRS Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems

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

September 14, 1994

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

Dear Chairman Selin:

SUBJECT: REVISED REGULATORY ANALYSIS GUIDELINES

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we discussed the proposed final "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." During this meeting, we had the benefit of discussions with representatives of the NRC staff. We note that the industry did not have the opportunity to review the staff response to public comments. We had provided comments on a preliminary version of these Guidelines to the Executive Director for Operations in a letter dated November 12, 1992. We also had the benefit of the documents referenced.

In our November 12, 1992 letter, we made a number of substantive comments on areas in which we disagreed with the staff proposals. In the revised version, the staff has satisfactorily addressed most of our earlier concerns. In addition, we believe the staff response to the public comments has been balanced and appropriate.

We believe these Guidelines will be valuable to the NRC staff in its various decision-making functions. At this time, we still have concerns in two areas:

1. Until new guidance has been developed on the appropriate monetary values to apply to adverse health and land contamination effects, the staff proposes the continued use of an undiscounted \$1000/man-rem. as a surrogate for the actual discounted values.

We do not support this proposal. The correct treatment requires separate, realistic values for each effect and these should be discounted for present-worth evaluation. The Guidelines should not be issued until a technically correct approach with the appropriate values is developed.

2. The revised Guidelines now propose a definition for containment failure that is "... consistent with the performance goal used in the review of evolutionary ALWRs and documented in SECY-93-087." This is a change from the definition used in a prior version of the Guidelines which was taken from NUREG-1150.

The definition in NUREG-1150, which addresses the risk dominant sequences, is the appropriate one for use in these Guidelines.

The issuance of the new Regulatory Analysis Guidelines should be delayed until these issues are reconsidered.

Sincerely,

T. S. Kress
Chairman

References:

1. Letter dated June 29, 1994, from C. J. Heltemes, Jr., NRC Office of Nuclear Regulatory Research, to T. S. Kress, ACRS Chairman, transmitting draft SECY Paper: Regulatory Analysis Guidelines of the U.S. NRC (Draft Predecisional)
2. Letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Revised Regulatory Analysis Guidelines

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

December 15, 1994

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

Dear Mr. Taylor:

SUBJECT: NRC TECHNICAL TRAINING PROGRAM

During the 416th meeting of the Advisory Committee on Reactor Safeguards, December 8-10, 1994, we discussed the NRC technical training program. Our Subcommittee on NRC Technical Training discussed this matter with representatives of the NRC staff during a meeting on December 7, 1994. In addition, two of our members toured the Technical Training Center (TTC) in Chattanooga, Tennessee on October 4, 1994. We also had the benefit of the documents referenced.

The TTC provides technical training in response to the needs identified by NRC program offices. Such training is limited to technical subjects and includes reactor technology, radiation protection, fuel cycle, safeguards, engineering support, and probabilistic risk assessment (PRA). Five full-scope simulators are maintained at TTC for training reactor inspectors, operator examiners, and others.

In discussions with Office for Analysis and Evaluation of Operational Data (AEOD) management and TTC representatives, we sensed a strong commitment to the evaluation and strengthening of existing programs, and a responsiveness to the emerging needs of the NRC in the areas of PRA and digital instrumentation and control systems.

Notwithstanding the broad nature of the existing inspector training program, we suggest that consideration be given to training related to water chemistry, health physics aspects of source terms, and the needs of inspectors monitoring licensee implementation of the Maintenance Rule (10 CFR 50.65). Additionally, the entire technical staff may benefit from training emphasis given to those aspects of reliability and uncertainty that are pertinent to the regulatory use of PRA and to performance-based regulation.

We plan to hold further discussions with the staff regarding training curricula in the areas of PRA and digital instrumentation

Mr. James M. Taylor

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and control systems. We plan to examine the staff's identification of learning objectives and its assessment of achieving individual course goals.

Sincerely,

T. S. Kress
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, "Technical Training Center Syllabus of Courses," 1994-1995
2. U. S. Nuclear Regulatory Commission, Office for analysis and Evaluation of Operational Data, "Technical Training Center Annual Report for Fiscal Year 1994"
3. Memorandum dated September 15, 1994, for All Employees from Kenneth A. Raglin, Technical Training Division, Office for Analysis and Evaluation of Operational Data, transmitting Technical Training Division Course Schedule for FY 1995

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

February 17, 1994

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

Dear Chairman Selin:

SUBJECT: SECY-93-270, "PROPOSED AMENDMENTS TO 10 CFR PART 73 TO PROTECT AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS"

During the 406th meeting of the Advisory Committee on Reactor Safeguards, February 10-11, 1994, we received further information about the proposed Rule from the NRC staff and from some public commenters, including NUMARC, the Nuclear Control Institute, and Mr. Scott Portzline, a private citizen.

This additional input has contributed to our understanding of the various positions, but appears not to have led to any mass reversal of positions on our parts. In addition, the staff has yet to propose a final version of the proposed Rule change, incorporating a response to the issues raised in the public comments. We plan to defer our own final judgment until that is done.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-270, dated September 29, 1993, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Proposed Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants
2. Letter dated January 3, 1994, from Thomas E. Tipton, NUMARC, to Samuel J. Chilk, Secretary of the Commission, Subject: Notice of Proposed Rulemaking -- "Protection Against Malevolent Use of Vehicles at Nuclear Power Plants," 58 Fed. Reg. 58804, November 4, 1993 Request for Comments
3. Letter dated January 3, 1994, from Daniel Horner and Paul Leventhal, Nuclear Control Institute, to Secretary of the Commission, regarding revision of NRC rule on protection against malevolent use of vehicles at nuclear power plants

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

February 17, 1994

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

Dear Mr. Taylor:

SUBJECT: THREE ISSUES RELATING TO THE 10 CFR PART 52 DESIGN
CERTIFICATION PROCESS FOR ALWRS

During the 406th meeting of the Advisory Committee on Reactor Safeguards, February 10-11, 1994, we discussed three issues that relate to the 10 CFR Part 52 design certification process for Advanced Light Water Reactors (ALWRs): (1) the staff's implementation of Reliability Assurance Program (RAP), (2) the staff's proposed use of "starred" Tier 2 Certified Design Material (CDM), and (3) Technical Specification requirements for onsite power sources for Evolutionary Light Water Reactors (ELWRs). We are commenting on these matters at this time because we believe that they need timely senior staff management attention. We had the benefit of the documents referenced.

ALWR Reliability Assurance Program

During our January 6-7, 1994 meeting, we heard a staff presentation on the RAP that is being required as a part of the design certification of ALWRs. The RAP requires both a design phase and an operational phase reliability assurance program (DRAP and ORAP). In addition, we reviewed your memorandum of August 2, 1993, in response to Commissioner Remick's questions on this subject. We also understand that OGC has concerns regarding the need for the DRAP and ORAP.

In our letter to you dated October 15, 1992, concerning "Proposed Guidance for Implementation of the Maintenance Rule," we noted that the RAP being required of ALWR COL holders ". . . will involve the establishment of a third kind of maintenance program (in addition to the maintenance programs required by the Maintenance Rule and the License Renewal Rule)." We suggested that consistent staff guidance was needed on the elements of an acceptable program that will satisfy these three sets of requirements. We have subsequently learned that a similar situation exists in the relationship between RAP and the quality assurance requirements of Appendix B to 10 CFR Part 50.

While we agree that PRA insights with respect to the reliability of risk significant structures, systems and components (SSCs) should be a part of maintenance and quality assurance programs for ALWRs, we continue to question the need for a separate RAP. We believe that senior staff management should perform a high level review of the need for the RAP. The objective of such a review should be to determine if it is possible to integrate those unique requirements of RAP that have a valid safety basis into the implementation of existing programs required for ALWRs (the Maintenance Rule, the License Renewal Rule, and Appendix B to 10 CFR Part 50).

The following aspects of RAP are of particular concern to us:

- The staff appears to believe that risk-significant SSCs should be given some sort of "special consideration" during the detailed design and procurement phases of an ALWR plant. It is not clear to us how the design engineering organization of a COL holder will be able to demonstrate that it has given "special consideration" to the procurement of risk-significant SSCs.
- The staff has not made it clear how the COL holder will develop reliability monitoring programs that will demonstrate that risk-significant SSCs are operated and maintained consistent with the PRA assumptions during the operational life of the plant. Demonstration of the reliability of risk-significant ALWR SSCs in any meaningful manner is clearly not feasible.

ALWR "Starred" Tier 2 Material

The staff has recently told us of its plan to designate certain Tier 2 CDM in the certification of the General Electric Nuclear Energy ABWR, and presumably in the certification of other ALWRs, as material which could not be changed by a COL holder under the 10 CFR 50.59 - like process, but would require prior review and approval by the staff. This will, in effect, create a three tier design certification process. Although there may be a valid need for this kind of restriction in certain cases, we recommend that senior staff management review each application of such "starred" Tier 2 CDM to ensure that the process is not being used in an arbitrary and capricious manner by the staff. In our view, the existing 10 CFR 50.59 - like process that a COL holder must use in order to change Tier 2 material generally provides the needed check and balance on changes to Tier 2 material.

ELWR Technical Specification Requirements for Onsite Power Sources

The staff informed us during our ABB-CE System 80+ Subcommittee meeting of December 8, 1993, that it is still considering Technical Specification requirements for onsite power sources for ELWRs. (A

February 17, 1994

similar, but somewhat different, issue exists with respect to the onsite power sources for the "passive" LWRs.) We have been interested for some time in the question of what credit will be given for the ELWR Alternate AC (AAC) source when one of the 1E Emergency Diesel Generators (EDGs) is out of service. Unlike the 1E EDGs, the AAC sources in the ELWR plant designs are not seismically qualified nor are they located within a structure hardened against the effects of tornados or hurricanes. This is particularly an important issue for the ABB-CE System 80+, where the onsite power sources consist of two 1E EDGs and a single AAC. If one of the 1E EDGs is out of service for maintenance, loss of offsite power (LOOP) would make the unit vulnerable to the single failure of the remaining 1E EDG under design basis accident conditions. Unless credit is given for the AAC (which may be damaged as a result of a seismic event or tornado or hurricane that caused the LOOP), the unit would have to be shut down whenever extended maintenance is performed on either of the 1E EDGs during power operation.

It appears to us that staff resolution of this matter is long overdue and that senior staff management attention to this issue is needed. Further, we believe that the Technical Specification Requirements for onsite power sources for ELWRs should be based on appropriate probabilistic risk considerations.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated August 2, 1993, from James M. Taylor, NRC Executive Director for Operations, to Commissioner Remick, Subject: SECY-93-087: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs
2. Report dated October 15, 1992 from David A. Ward, Chairman, ACRS to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.65
3. U.S. Nuclear Regulatory Commission, "Advance Copy of Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor Design," December 1993

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

April 13, 1994

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

Dear Chairman Selin:

SUBJECT: AMENDMENTS TO 10 CFR PART 73 TO PROTECT AGAINST
MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS

During the 408th meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 1994, we heard further presentations from the NRC staff on the rationale for the proposed rule change on the malevolent use of vehicles at nuclear power plants. Because of safeguards restrictions placed on us, the majority of the staff briefing was in closed session. In open session, we also heard brief statements by a private citizen and by a representative of the Nuclear Control Institute. In addition, a few of us recently visited an operating plant, and discussed the vehicle security situation. Finally, we have noted your March 28, 1994 response to our earlier letter dated December 10, 1993, and have seen the March 16, 1994 letter to you from Senators Baucus, et al., urging expeditious adoption of the new rule.

Your March 28 letter assured us that all relevant information will be considered by the Commission in coming to a final decision on the proposed rule, while emphasizing the difficulty, but not impossibility, of bringing quantitative considerations to bear on this subject.

The staff presentations were devoted to further analyses of certain conjectured scenarios, in support of the proposed rule, but we are constrained from discussing them in any detail here. We can say that (partly because of time limitations) the staff did not go beyond what was said at earlier meetings on the subject of vehicle bombs (a subject covered in our December 10, 1993, letter to you) but devoted its time to other uses of vehicles. Since the staff has made no changes of substance to the proposed rule since our earlier meetings, we are working with the same material as before, though somewhat enhanced by analysis of new scenarios.

The essence of risk-based regulation, to which the Commission is committed is that one should be thoughtful about implementing regulation for safety, and not necessarily do everything that seems at first blush reasonable. It is, for example, easy to think of

April 13, 1994

many actions that would very likely enhance plant security, some conceivably more than would vehicle barriers. One might require all chain-link fences to be of heavier gauge, or require that they be higher (which would also enhance protection against other threats), or that they be tripled, or that the pay of guards be doubled (to attract even higher quality candidates), or that psychological screening of employees be enhanced (remember the insider threat), and so forth. The point is that choices have to be made, and analyses should be made in order to come to those choices.

On balance, we have seen nothing new that alters either the majority or the minority views expressed in our December 10, 1993 letter. We said then that precipitous action might be justifiable if there were threat-related information to which we were not privy. We have been assured that there is not. Barring that, we continue to find little justification for a rush to judgment on this matter. If you wish us to continue to study this subject in even greater detail, we will be pleased to schedule another subcommittee meeting on an expedited basis.

An alternative available to the Commission might be to hold this proposed rule in abeyance, while the entire issue of plant vulnerability, including vehicle-borne explosives, the insider threat, and other possible intrusions, can be treated in an integrated way. They do interact, and it may be a mistake to treat them as independent.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I agree with the Committee conclusions, and also believe the staff proposal is in clear violation of the Backfit Rule. The analysis I had always thought to be required by the rule would have left the proposed rule changes wanting, had it been performed.

References:

1. Letter dated December 10, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants

2. Letter dated February 16, 1994, from Paul Leventhal, President Nuclear Control Institute, to J. Ernest Wilkins, Jr., ACRS Chairman, regarding the proposed truck-bomb rule
3. Letter dated March 16, 1994, from Max Baucus, Chairman, Senate Committee on Environment and Public Works, to Ivan Selin, NRC Chairman, regarding the adoption of a rule to require that nuclear power plants be protected against the use of vehicles for malevolent purposes
4. Letter dated March 28, 1994, from Ivan Selin, NRC Chairman, to J. Ernest Wilkins, Jr., ACRS Chairman, concerning the proposed rulemaking on protection against malevolent use of vehicles at nuclear power plants
5. Memorandum dated March 29, 1994, from Ashok Thadani, Associate Director for Inspection and Technical Assessment, NRR, to John Larkins, ACRS Executive Director, Subject: ACRS Review of Commission Paper on Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants

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

May 13, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE FOR SHUTDOWN AND LOW-POWER OPERATIONS

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we reviewed the NRC staff proposed Rule and associated Regulatory Guide pertaining to the conduct of shutdown and low-power operations. During this review, We had the benefit of discussions with representatives of the Office of Nuclear Reactor Regulation and the Office of the General Counsel, the Nuclear Energy Institute (NEI), and the Combustion Engineering Owners Group (CEOG). We have previously commented on the staff program to resolve this issue in our letters dated August 13, 1991, April 9, 1992, and September 15, 1992. We also had the benefit of the documents referenced.

In our September 15, 1992 letter, we commented on three issues that were of concern to us: proposed technical specifications for PWR containment integrity, proposed requirements for fire protection during shutdown, and the adequacy of the staff regulatory analysis. Your letter of October 16, 1992 indicated that the staff was in general agreement with our comments. (At the time of these letters, the staff was planning to utilize a generic letter, instead of rulemaking, to resolve this issue.) In addition, you stated that the staff would provide written responses to five questions raised by the Committee members during an April 1, 1992 Subcommittee meeting. The staff provided this information in a letter dated September 20, 1993, and we concluded that these responses were generally satisfactory.

Our present review has been based on the rulemaking package provided to the Committee to Review Generic Requirements (CRGR) for its review, as supplemented by a revised package containing changes the staff proposes to make in response to the recommendations made by the CRGR. In addition, we considered the views presented by the CEOG in its letter dated April 8, 1994.

The staff now proposes to resolve concerns regarding the conduct of shutdown and low-power operations by rulemaking that would require that licensees (1) plan and control outages in a way that provides

reasonable assurance that the key safety functions of maintaining the reactor subcritical, removing decay heat, and maintaining reactor coolant system (RCS) inventory will be preserved; (2) establish limiting conditions for operation and surveillance requirements for specific equipment relied on during shutdown and low-power operations; (3) demonstrate, by analysis, that those functions necessary to remove decay heat from the reactor can be maintained during cold shutdown and refueling conditions in the event of a fire in any plant area; (4) install instrumentation for monitoring water level in the RCS of pressurized water reactors during midloop operation.

We believe that improvements are needed in the conduct of shutdown and low-power operations. However, we have concluded that the staff has not made a sufficient case in its regulatory analysis either quantitatively or qualitatively to satisfy the requirements specified in 10 CFR 50.109. Where quantitative support for a backfit decision is not practicable, the use of subjective judgment should be acknowledged and the bases better substantiated than was done in this case.

Many of the staff-proposed improvements appear to have merit; some have already been adopted by the industry; others appear to require additional thought. (The CEOG provided us with data, for the period from 1989 through 1993, that demonstrate a substantial reduction in licensee events occurring during shutdown and involving loss of decay heat removal capability.) We believe that specific requirements of the Rule should continue to be the subject of a dialogue between the staff and NEI and that issuance of the Rule for public comment should be deferred until this dialogue is completed. We also believe that insights from the recently completed PRAs performed under a contract with the Office of Nuclear Regulatory Research should be considered.

Our comments relating to the safety improvements that the staff believes would result from this proposed rulemaking are as follows:

- In the regulatory analysis the staff states that "... a licensee program that (1) fully implements the guidelines in NUMARC 91-06 (Guidelines for Industry Actions to Assess Shutdown Management) and (2) incorporates the features regarding fire protection and instrumentation listed in Table 2.1 would be consistent with the staff assumptions regarding the administrative controls portion of this improvement (Improvement A)."

NEI believes that the industry initiative, as delineated in the NUMARC 91-06 document, obviates the need for including outage planning and control requirements in this rulemaking. NEI stated during our meeting that all power reactor licensees are implementing these Guidelines. The staff acknowledges

that implementation of these Guidelines has been "a significant and constructive step, effects of which have already been realized by many utilities ... in recent outages." We believe that past industry initiatives have proven to be an effective means of resolving safety issues without the need for rulemaking (e.g., Institute of Nuclear Power Operations accreditation of licensee training programs). This leads us to question the need for additional regulation relating to outage planning and control requirements.

- We do not believe that the staff has clearly defined what is expected of licensees relative to fire hazards assessment and associated fire contingency plans, including the bases for such plans. We plan to review the results of the NRC staff reassessment of its fire protection program as discussed in SECY-93-143. Discussion of shutdown fire hazards will be a part of this review.
- The staff has proposed a requirement for equipping PWRs with new water level instrumentation for midloop operation that would rely on measurement techniques not affected by pressure errors. The staff acknowledges that control of level, based on existing measurement techniques, has improved as a result of the requirements contained in GL 88-17, "Loss of Decay Heat Removal." The incremental safety improvement that would result from the addition of new water level instrumentation needs to be evaluated and contrasted with that resulting from more vigorous enforcement of the GL 88-17 requirements.
- The staff has proposed a number of technical specifications for the control of safety-related equipment during shutdown and low-power operations. NEI points out that these requirements overlap those cited in Section 50.65(a)(3) of the Maintenance Rule, which specifies that "In performing monitoring and preventive maintenance activities, an assessment of the total plant equipment taken out of service should be taken into account to determine the overall effect on the performance of plant safety functions." This section of the Maintenance Rule appears to provide the staff with the enforcement authority necessary to ensure proper control of safety-related equipment during shutdown and low-power operations. The use of such an approach also recognizes that the risk arising from shutdown and low-power operations is plant-specific in nature. Additionally, this approach would also provide licensees with more flexibility in their management of outage work.

We wish to be kept informed as development of this important issue progresses.

Sincerely,

T. S. Kress
Chairman

References:

1. Memo dated May 2, 1994, from M. Virgilio, Office of Nuclear Reactor Regulation, to J. Larkins, ACRS, transmitting revised copy of proposed Rule and associated draft Regulatory Guide on shutdown and low-power operations
2. Memorandum dated March 14, 1994, from F. Miraglia, Office of Nuclear Reactor Regulation, for E. Jordan, Chairman, Committee to Review Generic Requirements, transmitting proposed rulemaking package on shutdown and low-power operations containing: Federal Register Notice with proposed Rule, a draft Regulatory Analysis, draft Regulatory Guide 1.XXX, "Shutdown and Low-Power Operations at Nuclear Power Plants", and NUREG-1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States"
3. Letter dated April 8, 1994, from R. Burski, Chairman, CE Owners Group, to J. E. Wilkins, ACRS, transmitting comments on proposed regulatory requirements for shutdown and low-power operations
4. Letter dated March 28, 1994, from W. Rasin, Nuclear Energy Institute, to E. Jordan, AEOD, transmitting comments on proposed regulatory requirements for shutdown and low-power operations
5. Memorandum dated September 20, 1993, from A. Thadani, Office of Nuclear Reactor Regulation, for J. Larkins, ACRS, transmitting "Questions from the Operations Subcommittee Regarding Shutdown and Low-Power Operations"
6. Letter dated September 15, 1992, from D. A. Ward, Chairman, ACRS, to J. M. Taylor, EDO, Subject: NRC Staff's Proposed Resolution of Issues Identified in its Evaluation of Shutdown and Low-Power Operations
7. Letter dated October 15, 1992, from J. M. Taylor, EDO, to D. A. Ward, Chairman, ACRS, Subject: NRC Staff's Proposed Resolution of Issues Found During its Evaluation of Shutdown and Low-Power Operations
8. Letter dated April 9, 1992, from D. A. Ward, Chairman, ACRS, to J. M. Taylor, EDO, Subject: Evaluation of the Risks During Shutdown and Low-Power Operations for U.S. Nuclear Power Plants

9. Letter dated August 13, 1991, from D. A. Ward, ACRS Chairman, to J. M. Taylor, EDO, Subject: Evaluation of Risks During Low-Power and Shutdown Operations of Nuclear Power Plants

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

June 14, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSAL FOR MODIFYING THE NRC RULEMAKING PROCESS

During the 409th and 410th meetings of the Advisory Committee on Reactor Safeguards, May 5-7 and June 9-10, 1994, we discussed SECY-94-141 which proposes extensive revision of the rulemaking process superseding the traditional method in several key areas. This SECY paper is in response to the several initiatives to streamline the rulemaking process throughout the Federal Government and within the NRC. During the May meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We find that the provisions of SECY-94-141 address shortcomings of the present rulemaking process. We believe that the proposed revision of the rulemaking process is an overdue step in the right direction and urge its approval for implementation. We will work with the EDO to revise our Memorandum of Understanding to reflect these provisions.

Sincerely,

T. S. Kress
Chairman

References:

1. SECY-94-141 dated May 23, 1994, from William C. Parler, General Counsel and James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Improvement of the Rulemaking Process
2. Memorandum dated December 15, 1993, from William C. Parler, General Counsel, for the Commissioners, Subject: Industry Suggestions for Streamlining NRC Rulemaking Procedures

3. Letter dated December 7, 1993, from Joe F. Colvin, Nuclear Management and Resources Council, for The Honorable Ivan Selin, NRC Chairman, Subject: NRC Rulemaking Process
4. Memorandum of Understanding dated May 19, 1988, between the Advisory Committee on Reactor Safeguards - ACRS Chairman and the Nuclear Regulatory Commission Staff - EDO, Subject: ACRS Participation in the Development of NRC Rules, Policy Matters, and Safety-Related Guidance

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

September 19, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED REVISIONS TO APPENDIX J TO 10 CFR PART 50,
"PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR
WATER-COOLED POWER REACTORS"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we reviewed the proposed revisions to Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Our Subcommittee on Containment Systems also reviewed this matter at a meeting on September 7, 1994. During this review, we had the benefit of discussions with representatives of the NRC staff, Nuclear Energy Institute (NEI), Grand Gulf Nuclear Station (Entergy Operations, Inc.), and ANS-56.8 Working Group (Containment System Leakage Testing Requirements). We also had the benefit of the documents referenced.

We are in general agreement with the proposed revisions to Appendix J and have no objection to the publication of the proposed rule for public comment. The changes proposed do not appear to have significant potential to increase public risk and, in fact, may reduce risk by decreasing the probability of accidents during shutdown. In addition, the changes will permit staff and industry resources to be redirected to more risk-significant issues.

The staff identified two issues that remain unresolved with industry. These are: (1) the proposed rule allows a maximum interval for leakage testing of Type C components (isolation valves) of 60 months, whereas industry would prefer a staggered test program leading to a maximum of 120 months; and (2) the staff proposes that certain leak testing provisions be incorporated into the technical specifications for the individual plants, whereas the industry proposes that the leak testing provisions be a commitment in the Final Safety Analysis Report (FSAR).

With regard to the leakage testing interval for Type C components, the arguments for the 120-month interval are reduction in costs, in occupational exposure, and in shutdown risks. The staff arguments

for an initial 60-month limit are: (1) a conservative approach should be adopted until experience is gained, and (2) aging effects on leakage may escape timely detection if a period longer than 60 months is allowed. We accept the staff position on this issue, which includes the option for a 120-month interval after evaluating experience with the proposed rule. Our acceptance is conditional on the assumption that valve operability (as opposed to leakage) will be demonstrated appropriately by other means such as those already implemented under Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

We note that any shutdown risk benefit that may be gained by increasing the test interval has not been quantified. In addition, the staff has acknowledged that it has not looked for aging effects on valve leakage in older plants. We recommend that the staff examine both of these issues in order to provide additional insights relative to the appropriate maximum test interval for Type B and C components. The shutdown risk issue could be evaluated by extension of the recently completed shutdown risk assessments for Surry and Grand Gulf nuclear plants.

With respect to the second unresolved issue, both the staff and NEI agree that the allowable leakage rate for the containment (which we view as the performance goal) should be included in the Technical Specifications (TS). The staff is still considering requirements that may be needed in the TS to ensure that program changes are reviewed by the staff. An example is the algorithm to be used for extension of Type C isolation valve leakage testing. NEI argues that it is sufficient to place these requirements in the FSAR so that changes can be made using the 10 CFR 50.59 process. Since the additional TS requirements proposed by the staff are counter to the concept of the performance-based Maintenance Rule, we recommend that the staff adopt the NEI position on this issue.

We plan to review this matter after reconciliation of the public comments.

Additional comments by ACRS Members Thomas S. Kress and Robert L. Seale and ACRS Members James C. Carroll, Ivan Catton, and William J. Lindblad are presented below.

Sincerely,

T. S. Kress
Chairman

Additional Comments by ACRS Members Thomas S. Kress and Robert L. Seale

We fully agree with the Committee that there is unlikely to be an unacceptable increase in risk as a result of this proposed change to the leakage testing interval and that this is an appropriate area to provide some regulatory relief for the industry. Nevertheless, we have two objections to the form of the proposed revisions:

1. We believe a bad precedent is set for performance-based regulations by having the relaxation (or tightening) of the regulatory oversight be on the performance measure frequency itself. It should be a general principle that these be separate.
2. We are unconvinced that an adequate technical basis has been established that two consecutive successful leakage tests provide appropriate criteria for acceptable performance in this case. This, again, sets a bad precedent for supposedly performance-based regulations.

Additional Comment by ACRS Members James C. Carroll, Ivan Catton, and William J. Lindblad

While we believe the Appendix J revisions proposed by the staff will protect public health and safety, the further provisions that were proposed by NEI (staggered testing of classes of Type C components with a maximum testing interval of 120 months) seem to us to be proper as well. The conditions under which extended test intervals would be permitted appear to be consistent with those contemplated by the Maintenance Rule.

References:

1. Memorandum dated August 23, 1994, from Joseph A. Murphy, Office of Nuclear Regulatory Research, NRC, for John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Performance-Based Containment Leakage Test Rulemaking (Transmitting Draft SECY Paper for the Commissioners from James M. Taylor, EDO, undated)
2. Nuclear Energy Institute, NEI 94-01, Draft Revision C, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," August 1, 1994
3. U.S. Nuclear Regulatory Commission, NUREG-1493, Draft (Revision 2, 3/31/94), "Performance-Based Containment Leak-Test Program"
4. Electric Power Research Institute/Science Applications International Corporation, EPRI TR-104285, Final Report dated August 1994, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals"

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

December 19, 1994

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

Dear Chairman Selin:

SUBJECT: REVISIONS TO 10 CFR PART 71, PACKAGING AND TRANSPORTATION
OF RADIOACTIVE MATERIAL

During the 416th meeting of the Advisory Committee on Reactor Safeguards, December 8-10, 1994, we discussed the subject proposed final rule with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the document referenced.

The staff stated that the proposed revisions are being made for two reasons:

- to make U.S. transportation regulations compatible with the 1985 edition of the International Atomic Energy Agency (IAEA) regulations, and
- to promulgate new criteria for air shipment of plutonium as required by statute.

The following are the proposed revisions for the U.S. regulations:

- requiring additional hypothetical accident test criteria for certain types of packages,
- increasing the number of radionuclides with A_1 and A_2 quantities that determine shipping container requirements,
- changing the A_1 and A_2 quantities for some radionuclides,
- simplifying the fissile material transport classes,
- revising requirements for shipment of "low specific activity" (LSA) material, and
- including the criteria for packages used to transport plutonium.

The Committee supports the concept of making the U.S. regulations on packaging and transport of radioactive materials compatible with IAEA regulations if this can be done without undue compromise of safety. In the past, the ACRS has extensively reviewed the safety aspects of the existing regulations. Our present review has not been in-depth because it is apparent that the proposed revisions have minor safety significance.

Our concerns are not with the revisions themselves but with the associated regulatory process that strikes us as being somewhat atavistic for the following reasons:

- the proposed revisions are solely developed deterministically and do not have a clear technical risk basis,
- a probabilistic risk analysis is lacking,
- a regulatory analysis for the departures from the IAEA regulations appears to be incomplete, and
- apparently, there have been no interactions with industry since 1989.

The IAEA regulation for LSA and surface-contaminated-object material calls for a limit on the exposure level at a particular distance from the unshielded material. The staff is concerned that large quantities of resin beads shipped in LSA containers could change geometry and lose self-shielding during an accident. As a result, personnel exposure could be greater than originally analyzed. Therefore, the staff proposes to depart from the IAEA regulations by placing a limit on the quantity of activity that can be shipped in LSA packages. We believe this departure would fail a cost/benefit screen as well as a screen on substantial increase in safety. We believe that a well-founded regulatory analysis that properly considers the probability and level of the greater exposure, and the practical limits on the mass of material in a shipment would indicate that the safety benefit would not justify the burdens created.

We recommend that the proposed revisions to 10 CFR Part 71, with the exception of the plutonium air shipment provisions, be reevaluated with the objective of making them equivalent to the IAEA regulations. We also recommend that a risk analysis be performed for the purposes of understanding the risk profile and quantifying the safety margins. If departures from IAEA regulations are found to be necessary based on risk considerations, dialogue should be renewed with those in the industry likely to be

The Honorable Ivan Selin

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affected. We encourage a closer and continuing interaction with licensees in the consideration of these issues.

Sincerely,

A handwritten signature in dark ink, appearing to read "T. S. Kress", with a stylized, flowing script.

T. S. Kress
Chairman

Reference:

Draft SECY, undated, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Final Rule on Revision of NRC Transportation Regulations (received November 8, 1994)

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

September 12, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED GENERIC LETTER 94-XX, "VOLTAGE-BASED REPAIR
CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES"

During the 412th meeting of the Advisory Committee on Reactor Safeguards, August 4-5, 1994, we reviewed the subject generic letter (GL), an associated differing professional opinion (DPO), and a draft of an Advance Notice of Proposed Rulemaking on Steam Generator Tube Integrity. During the 413th meeting, September 8-10, 1994, we discussed the NRC staff's revised calculations for radiological consequences of a main steamline break associated with a degraded steam generator. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), as well as the author of the DPO. We also had the benefit of the documents referenced. In part, this report is in response to a request made by the Executive Director for Operations in a July 15, 1994, memorandum to the Executive Director of the Advisory Committee on Reactor Safeguards.

Although existing mechanics-based design criteria and evaluation methods have served to ensure adequate steam generator tube integrity, they appear to be overly conservative for some types of degradation, and result in unnecessary tube plugging or repair. The proposed GL provides an alternate approach applicable solely to axially oriented outside diameter stress corrosion cracking (ODSCC) of tubes at the tube-support-plate intersections in Westinghouse steam generators with drilled-hole support plates.

We support the issuance of the proposed GL for public comment. We have reviewed the DPO and do not believe that it identifies any fundamental shortcomings in the approach proposed in the GL.

The DPO cites a high core damage frequency (CDF) of 3.4×10^{-4} /RY. This value was based on a preliminary scoping analysis performed by the Office of Nuclear Regulatory Research (RES). Subsequent analyses performed by RES in support of the application of the interim plugging criteria for the Trojan Nuclear Plant and for NUREG-1477 give CDFs of less than 2×10^{-6} /RY. These values are based on conservative estimates of leakage from degraded tubes. Except perhaps for steamline breaks, the structural restraint provided by the tube-support plate provides a high degree of assurance against tube bursts.

The criticism in the DPO of the approach used in the proposed GL and in the Standard Review Plan to compute radiological releases during a main steamline break appears to warrant further consideration. The basis for the definition of the iodine spike during a rapid depressurization transient as 500 times the equilibrium release rate is not clear. However, an alternate calculation of the release based on the gap inventory of iodine in leaking fuel elements appears to give comparable releases. In both approaches there appears to be margin in meeting the 10 CFR Part 100 limits. The staff should review the spiking data or consider other approaches to estimate the iodine release to provide a more satisfactory basis for the radiological dose estimates. In particular, we encourage the staff to quantify the level of conservatism in its analyses.

While the proposed GL appears to provide a useful interim approach for assessing steam generator tube integrity, the database for the present empirical correlations for burst pressure and leakage with the bobbin coil voltage, appears to be only marginally adequate, and more data need to be developed.

The use of such empirical correlations as the basis for assuring the integrity of steam generator tubing would also seem to require an ongoing tube-pull program with associated burst and leak testing and metallurgical examinations as outlined in the proposed GL to ensure that the correlations remain valid as degradation continues. In the longer term, it would be worthwhile to reconsider a fracture-mechanics-based approach utilizing improved non-destructive examination techniques that provide more accurate detection and characterization of degradation. Ongoing efforts in RES and in industry to develop and implement such an approach should be continued and encouraged.

We agree with the staff position that rulemaking is the preferred regulatory approach to the problem of steam generator tube degradation, although we are skeptical that a new rule can be developed as expeditiously as the proposed schedule suggests. The overall objective and attributes of the new rule, as described by the staff, pay proper obeisance to performance-based regulation. We would like to be kept informed of the progress by the staff in the implementation of a performance-based approach.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated July 8, 1994, from F. J. Miraglia, Deputy Director, Office of Nuclear Reactor Regulation, for E. L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: CRGR Review of Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. Memorandum dated July 15, 1994, from J. M. Taylor, NRC Executive Director for Operations, for J. T. Larkins, ACRS Executive Director, Subject: ACRS Review of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes
3. U.S. Nuclear Regulatory Commission, 10 CFR Part 50, RIN 3150-, Steam Generator Tube Integrity (7590-01), Draft Advance Notice of Proposed Rulemaking, received July 20, 1994
4. Memorandum dated August 17, 1994, from J. A. Calvo, NRC Office of Nuclear Reactor Regulation, for J. T. Larkins, ACRS Executive Director, Subject: Revisions to Slides Used by Staff During August 3, 1994, Subcommittee Briefing on Steam Generator Alternate Repair Criteria
5. U.S. Nuclear Regulatory Commission, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," Draft Report for Comment, June 1993
6. Memorandum dated January 15, 1993, from E. S. Beckjord, Director, Office of Nuclear Regulatory Research, to T. E. Murley, Director, Office of Nuclear Reactor Regulation, Subject: Interim Plugging Criteria for Trojan Nuclear Plant

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

February 23, 1994

The Honorable Albert Gore, Jr.
President of the United States Senate
Washington, D.C. 20510

Dear Mr. President:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as had been provided before 1986.

In 1993, we reviewed various NRC activities, several of which included significant research elements directed to the reduction of uncertainties in the present knowledge base. Enclosed are copies of the reports that we have provided to the NRC during the past year on these matters. We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

*Enclosures:

1. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements, February 19, 1993
2. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, March 18, 1993
3. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Human Performance in Operating Events, March 19, 1993

4. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Definition of a Large Release for Use With Safety Goal Policy, April 22, 1993
5. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 26, 1993
6. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Review of Organizational Factors Research Program, April 27, 1993
7. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals, May 26, 1993 (Revised June 16, 1993)
8. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Proposed Draft Regulatory Guides, DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," and DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," July 15, 1993
9. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Proposed Rule Amending Fracture Toughness Requirements for Light Water Reactor Pressure Vessels, Proposed Rule Regarding Requirements for Thermal Annealing of Reactor Pressure Vessels, and Draft Regulatory Guide on Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels, September 20, 1993
10. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, November 16, 1993
11. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification, November 18, 1993
12. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Thermo-Lag Fire Barriers, December 16, 1993
13. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Diversity of the Method of Measuring Reactor Pressure Vessel Water Level in the Advanced and Simplified Boiling Water Reactor Designs, December 16, 1993

*For Items 1 through 13, see NUREG-1125, Volume 15, 4/94.

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

February 23, 1994

The Honorable Thomas S. Foley
Speaker of the United States
House of Representatives
Washington, D.C. 20515

Dear Mr. Speaker:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as had been provided before 1986.

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Sincerely,

J. Ernest Wilkins, Jr.
Chairman

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8. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Proposed Draft Regulatory Guides, DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," and DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," July 15, 1993
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*For Items 1 through 13, see NUREG-1125, Volume 15, 4/94.

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

September 20, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL VERSION OF NUREG-1465, "ACCIDENT SOURCE
TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we discussed the proposed final version of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." During the meeting, we had a discussion with the staff regarding how comments on the draft version of this document have been accommodated in the final version. We also had a presentation by a representative of Northeast Utilities on the safety importance of adopting proposed accident source term timing assumptions. The draft version was discussed with the Committee at the 381st meeting in January 1992, and comments were provided in our report dated January 15, 1992. We also had the benefit of the documents referenced.

NUREG-1465 defines accident source terms for use in the safety analysis of future light water reactors to replace the source term specified in Regulatory Guides 1.3 and 1.4. The proposed source terms are based on the vast amount of research sponsored over the last 15 years by the NRC and others. The proposed source terms specify the releases of eight categories of radionuclides over four time intervals after the initiation of an accident. Most of these radionuclides are expected to form aerosol particles in the containment. Only the noble gases and 5 percent of the iodine are in gaseous form. This contrasts with the source term now used which specifies an instant release consisting of 100 percent of the core inventory of the noble gases and 50 percent of the iodines (half of which are assumed to deposit on interior surfaces very rapidly) to the containment.

We believe it is important to have more realistic accident source terms available for regulatory activities. NUREG-1465 presents source terms which are a vast improvement over the source term now available. We do, however, have some comments.

A variety of calculations has been examined to develop the proposed source terms. In some cases, bounding values determined from these calculations have been adopted. In other cases, mean values have been selected, and in still others, values less than the mean have been chosen. As a result, it is difficult to ascertain the conservatism inherent in the proposed source terms. We believe it important to clarify this level of conservatism especially since the proposed source terms may be used for the analyses of both design basis and beyond design basis accidents. Appropriate levels of conservatism are quite different for these two classes of accidents.

Release fractions of some categories of radionuclides have been adjusted in the final version of NUREG-1465 from values in the draft that were derived from calculations. It appears that these adjustments have been based on expert opinions provided in comments by reviewers of the draft report. We believe these adjustments need to be better justified or not be made.

Ongoing source term research activities may yield results that would substantially alter the understanding that has been the basis of the proposed source terms. A mechanism is needed for timely updating of regulatory source terms in response to significant research findings.

The target application of the proposed source terms is to future light water reactors. Since the source terms have been derived from calculations for existing light water reactors, explicit provisions should be included in NUREG-1465 to accommodate specific features of future reactors.

We agree that licensees of existing reactors should not be required to adopt the proposed source terms. Information provided to the Committee suggests that use of realistic timing assumptions for radionuclide releases to the containment during accidents can lead to safety improvements in existing plants. We urge that the risk implications be evaluated and consideration be given to allowing current licensees the option of using the timing assumptions in the proposed source terms without performing a complete source term reanalysis.

We emphasize the importance of realistic source terms in regulatory applications and believe that the use of realistic source terms could result in changes in reactor design and operation that reduce risk. We continue to be interested in the future application of the proposed source terms to specific regulatory areas and issues and wish to be kept informed.

Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding this matter.

Sincerely,

A handwritten signature in dark ink, appearing to read "W. J. Lindblad", with a stylized, flowing script.

W. J. Lindblad
Vice-Chairman

References:

1. Memorandum dated August 5, 1994, from Themis P. Speis, RES, for John T. Larkins, ACRS, transmitting Draft Final NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"
2. Letter dated April 29, 1994, from J. F. Opeka, Connecticut Yankee Atomic Power Company/Northeast Nuclear Energy Company, to Mr. W. T. Russell, Director, NRR, Subject: Accident Source Term Timing Assumptions
3. Report dated January 15, 1992, from David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, to Ivan Selin, Chairman, NRC, Subject: Proposed 10 CFR Part 50 and Part 100 (Nonseismic) Rule Changes and Proposed Update of Source Term

BIBLIOGRAPHIC DATA SHEET

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This compilation contains 30 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1994. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U. S. Library of Congress. The reports are categorized by the most appropriate generic subject area and by chronological order within subject area.

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