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NOTE TO EDITORS:

The Nuclear Regulatory Commission has received the attached letter-type report from its independent Advisory Committee on Reactor Safeguards that provides comments on safety aspects of ABB-Combustion Engineering's application for certification of its advanced reactor design designated System 80+.

In addition, the ACRS has sent two letter reports to the NRC's Executive Director for Operations. The letters provide comments on a proposed rule for shutdown and low-power operations and a draft policy statement on the use of probabilistic risk assessment methods in reactor regulatory activities.

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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

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 (<u>Guidelines for Industry Actions to Assess Shutdown Management</u>) 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

<u>References</u>:

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

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