NUREG-1125
Volume 12

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

1990 Annual

U.S. Nuclear Regulatory Commission

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April 1991

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ABSTRACT

This compilation contains 31 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1990. 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 divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subject. Part 1 contains ACRS reports alphabetized by project name and by chronological order within project name. Part 2 categorizes the reports 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 1990. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
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

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	Dr. Paul G. Shewmon Ohio State University
	Dr. Chester P. Siess, Retired (Prof. Emeritus) University of Illinois
	Mr. David A. Ward, Retired E. I. du Pont de Nemours & Company Savannah River Laboratory
	Dr. J. Ernest Wilkins, Jr. Clark Atlanta University

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

November 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SECY-90-353, LICENSING REVIEW BASIS DOCUMENT FOR THE COMBUSTION ENGINEERING, INC. SYSTEM 80+ EVOLUTIONARY LIGHT WATER REACTOR

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we reviewed the staff's SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990. Our Subcommittee on Advanced Pressurized Water Reactors also considered this matter during a subcommittee meeting on November 1, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of Asea Brown Boveri Combustion Engineering. We also had the benefit of the documents referenced.

The staff has recommended that the Licensing Review Basis (LRB) effort for the Combustion Engineering (CE) System 80+ design, which is well advanced, be continued to completion. There does not appear to be any substantive disagreement between the staff and CE on issues addressed in the LRB document.

The only approved LRB document was proposed by the General Electric Company (GE) as a way of obtaining early agreement with the staff on major process and technical issues for the review of its advanced boiling water reactor design certification application. It was approved by the Director of NRR in a letter to Mr. R. Artigas, GE, on August 7, 1987. This letter contains the qualification that the LRB represented the approach in "certain key areas" that GE was committed to follow ". . . until final Commission positions and staff requirements are defined and implemented." At that time, neither 10 CFR Part 52 nor Commission-approved staff positions relating to the certification of advanced light water reactors such as SECY-90-016 (referenced) were available. We note that 10 CFR Part 52 does not discuss the use of LRB documents as a part of the final design approval or certification process. These regulatory requirements and others under development have preempted the need for and diminished the usefulness of an LRB document for the CE System 80+ design. We recommend that no further effort be devoted to the proposed LRB document for the CE System 80+ design.

Additional comments by ACRS members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr., are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr.

We understand that this LRB document can be completed and issued with relatively little additional effort. If so, we would prefer to see an orderly disposition of this LRB document in accordance with the staff recommendation in SECY-90-362 (referenced). We would agree with our colleagues that the CE System 80+ LRB effort be terminated now if the Commission, the staff, and the ACRS need to invest any significant additional effort.

References:

1. SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990.
2. SECY-90-362, "Staff Comments on the Continuing Need for a License Review Basis Document for Each Passive Design," dated October 24, 1990.
3. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," dated January 12, 1990.
4. Letter LD-90-005 dated January 22, 1990 from A. E. Scherer, Combustion Engineering, to R. Singh, Subject: System 80+ Licensing Review Basis Document.
5. Letter LD-90-060 dated August 28, 1990, from E. H. Kennedy, Combustion Engineering, to Thomas V. Wambach, NRC, Subject: Licensing Review Basis for the System 80+ Standard Design.

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

December 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: FULL-TERM OPERATING LICENSE FOR THE DRESDEN NUCLEAR POWER STATION, UNIT 2

During the 368th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 1990, we completed our review of the application by the Commonwealth Edison Company (licensee) for conversion of the provisional operating license (POL) for the Dresden Nuclear Power Station, Unit 2, to a full-term operating license (FTOL). During our review, we had the benefit of discussions with representatives of the licensee and the NRC staff. We also had the benefit of the documents referenced. The Committee most recently discussed and reported on this plant in a letter dated December 13, 1982, relating to the Systematic Evaluation Program (SEP) review of Dresden, Unit 2.

Dresden, Unit 2, received a POL in December 1969 and began commercial operation in July 1970. The licensee applied for an FTOL in November 1972, but review of this application was deferred by the NRC staff in 1975, along with several other FTOL reviews. In 1978, Dresden, Unit 2, was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP. We call attention to the fact that Dresden, Unit 3, was given an FTOL in January 1971, after a rule change had eliminated the POL as an option. Units 2 and 3 are essentially identical.

The Committee, in its December 13, 1982 letter reporting on the results of the SEP as applied to Dresden, Unit 2, indicated that its review of the FTOL would be deferred until the NRC staff had completed its actions on the SEP issues that were still pending, and on the Unresolved Safety Issues (USIs) and TMI Action Plan items. All but three of the SEP issues were resolved to the satisfaction of the NRC staff in the manner reported in Supplement 1 to the Integrated Plant Safety Assessment Report for Dresden, Unit 2. The status of these three issues and of the USI and TMI Action Plan items has been discussed by the staff in its Safety Evaluation Report related to the FTOL for Dresden, Unit 2. We believe that the procedures and schedules that have been agreed to for the resolution of these items are satisfactory, and that the remaining actions to resolve these items would not be accelerated by withholding an FTOL.

We believe that there is reasonable assurance that the Dresden Nuclear Power Station, Unit 2, can continue to be operated at power levels up to 2527 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1403, "Safety Evaluation Report Related to the Full-Term Operating License for Dresden Nuclear Power Station, Unit 2," dated October 1990
2. U. S. Nuclear Regulatory Commission, NUREG-0823, Supplement No. 1, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, dated October 1989

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

February 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED POWER LEVEL INCREASE FOR INDIAN POINT NUCLEAR GENERATING STATION UNIT 2

During the 358th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1990, we reviewed the application of Consolidated Edison Company of New York (Licensee) for a license amendment, to permit it to operate the Indian Point Nuclear Generating Station Unit 2 at a core thermal power level up to 3071.4 MWt. The current core power level limit is 2758 MWt, so this is approximately an 11 percent increase. This matter was discussed by our Subcommittee on the Systematic Assessment of Experience, on February 6, 1990. During these meetings, we had the benefit of discussions with representatives of both the NRC staff and the Licensee. We also had the benefit of the documents referenced. The NRC staff recommends approval of this application.

The plant was originally licensed in 1973, at a core thermal power level up to 2758 MWt, though the original analyses and supporting environmental assessments, with the exception of the emergency core cooling system (ECCS), were made for a core thermal power level of 3216 MWt. The ECCS was evaluated at 2758 MWt. There is nothing in the history to suggest that the lower power level of the original license was based on anything other than a (commendable) caution, since this was the first of the large Westinghouse 4-loop plants to seek a license. Since this is a license amendment, the staff review is based on the original license requirements, and our review is confined to the implications of the proposed power level increase, not to a review of the original license decision.

Since nearly all the original analyses were performed at the higher power, the remaining need was to demonstrate ECCS operability at the proposed power, and this was done in May of 1989. The analyses were reviewed by the NRC staff, and found to be in compliance with

10 CFR 50.46 and Appendix K, with suitable conservatism. We have no reason to question these conclusions.

Many new requirements, not all due to the Three Mile Island accident, have been levied since the original license was issued in 1973. Some of these are power related, and the staff should assure itself that those will be met at the new power level. The Licensee has assured the NRC, in a letter dated February 8, 1990, that that is the case.

Subject to resolution of this matter to the satisfaction of the NRC staff, we believe that the Indian Point Nuclear Generating Station Unit 2 can be operated at core power levels up to 3071.4 MWt without undue risk to the health and safety of the public.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>References</u>:

1. Memorandum dated January 29, 1990 from S. A. Varga, Nuclear Regulatory Commission, to R. F. Fraley, ACRS, Subject: Transmittal of Revision to Draft Safety Evaluation to Increase Licensed Thermal Power Level of the Indian Point Nuclear Generating Unit No. 2
2. Letter dated September 30, 1988 from S. Bram, Consolidated Edison Company to U. S. Nuclear Regulatory Commission transmitting Application for Amendment to Operating License (Indian Point Station Unit 2)
3. Letter dated February 8, 1990 from S. Bram, Consolidated Edison Company to Donald S. Brinkman, Nuclear Regulatory Commission, Subject: Application for License Amendment to Increase Authorized Power Level

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

December 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: FULL-TERM OPERATING LICENSE FOR THE PALISADES NUCLEAR PLANT

During the 368th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 1990, we completed our review of the application by the Consumers Power Company (licensee) for conversion of the provisional operating license (POL) for the Palisades Nuclear Plant to a full-term operating license (FTOL). Our Subcommittee on FTOL Conversions also discussed this matter during a meeting on December 5, 1990. During our review, we had the benefit of discussions with representatives of the licensee and the NRC staff. We also had the benefit of the documents referenced. The Committee most recently discussed and reported on this plant in a letter dated May 11, 1982, relating to the Systematic Evaluation Program (SEP) review of Palisades.

The Palisades Nuclear Plant received a POL in December 1969 and began commercial operation in March 1971. The licensee applied for an FTOL in January 1974, but review of this application was deferred by the NRC staff in 1975, along with several other FTOL reviews. In 1978, Palisades was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP.

The Committee, in its May 11, 1982 letter reporting on the results of the SEP as applied to Palisades, indicated that its review of the FTOL would be deferred until the NRC staff had completed its actions on the SEP issues that were still pending, and on the Unresolved Safety Issues (USIs) and TMI Action Plan items. All but three of the SEP issues were resolved to the satisfaction of the NRC staff, as reported in Supplement 1 to the Integrated Plant Safety Assessment Report for Palisades. The status of these three issues and of the USIs and TMI Action Plan items has been discussed by the staff in its Safety Evaluation Report related to the FTOL for Palisades. We believe that the procedures and schedules that have been agreed to for the resolution of these items are satisfactory, and that the remaining actions to resolve those items would not be accelerated by withholding an FTOL at this time.

We believe that there is reasonable assurance that the Palisades Nuclear Plant can continue to be operated at power levels up to 2530 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1424, "Safety Evaluation Report Related to the Full-Term Operating License for Palisades Nuclear Plant," dated November 1990
2. U.S. Nuclear Regulatory Commission, NUREG-0820, "Integrated Plant Safety Assessment, Systematic Evaluation Program - Palisades," dated October 1982

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

December 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: WESTINGHOUSE'S APPLICATION FOR PRELIMINARY DESIGN APPROVAL FOR THE RESAR SP/90 DESIGN

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we completed our review of Westinghouse's application for Preliminary Design Approval (PDA) for the Westinghouse Reference Safety Analysis Report (RESAR SP/90) nuclear power block (NPB). We heard presentations from the NRC staff and the applicant concerning the staff's draft Safety Evaluation Report (SER) (NUREG-1413) for this PDA during our meeting. Representatives of the staff and of the Office of the General Counsel (OGC) discussed the related draft PDA document. Our Subcommittee on the Advanced Pressurized Water Reactors has held a series of meetings with the staff and representatives of the applicant regarding this matter over the past two and a half years. We also had the benefit of the documents referenced.

1.0 <u>Scope and History of RESAR SP/90 Application</u>

The RESAR SP/90 is an evolutionary (as contrasted with passive) Advanced Light-Water Reactor (ALWR) design for a single-unit NPB, rated at a reactor power of 3800 MWt. Although many basic design decisions were made by Westinghouse prior to completion of the EPRI ALWR Utility Requirements Document, the design of this four-loop pressurized water reactor generally conforms to the EPRI requirements for such designs.

RESAR SP/90 NPB contains preliminary design information for the portion of the design that encompasses NPB buildings, structures, systems, and components. Specifically excluded from the scope are the turbine building, the waste disposal building, the service building, the administration building, the service water/cooling water structure, and the ultimate heat sink. These features will be the design responsibility of an applicant proposing to build a facility referencing the RESAR SP/90 design. Interface information addressing the pertinent safety-related design requirements necessary to ensure the compatibility of the referenced system with the plant-specific portion of the facility has been included in the RESAR SP/90 application.

On October 24, 1983, Westinghouse submitted an application for a PDA for RESAR SP/90 NPB design in accordance with 10 CFR Part 50, Appendix O, "Standardization of Design: Staff Review of Standard Designs," which was the then existing regulatory basis for this type of application. The application was docketed on November 30, 1983 (Docket No. 50-601). The RESAR SP/90 application describing the design of the NPB was submitted in modular form during the period from October 23, 1983 to March 9, 1987. In addition, the information in RESAR SP/90 has been supplemented by 47 amendments to these modules.

2.0 Regulatory Background

Before the promulgation of 10 CFR Part 52 in May of 1989, the review of RESAR SP/90 had been performed by the staff pursuant to Appendix O to 10 CFR Part 50, using a procedure similar to that used for custom plant reviews for which guidance to staff reviewers is provided in the Standard Review Plan. This evaluation was analogous to a construction permit (CP) licensing review for a specific facility and conducted with the intent that, following satisfactory completion of the reviews performed by the staff and the ACRS, a PDA could be issued by the staff. The promulgation of 10 CFR Part 52 resulted in the transfer of Appendix O to 10 CFR Part 52; hence a PDA can now be issued for this application pursuant to 10 CFR Part 52. A PDA is optional for a Final Design Approval (FDA) and/or Design Certification under the provisions of 10 CFR Part 52.

3.0 The Staff's SER and the PDA

The SER and PDA represent the first stage of the staff's review of the design, construction, and operation of the RESAR SP/90 design. During our meetings, we learned that there is no prospective CP applicant nor does Westinghouse intend to apply for an FDA and/or Design Certification of the RESAR SP/90 design until there is a proven interest on the part of a domestic or foreign utility. The staff's SER summarizes the results of the staff's radiological safety review of the RESAR SP/90 NPB design and delineates the scope of the technical details considered in evaluating the proposed design. This review took place over the period of October 1983 to October 1989 (the date on which the staff decided to close its review). Environmental aspects were not considered in the staff review of RESAR SP/90, but would be addressed in a utility's plant-specific application.

3.1 <u>Comments on the Staff's SER</u>

There are 170 open items that will require resolution during the review of a plant-specific application for an Operating License (OL). Most of these appear to be the kind of open issues expected at this stage of the design. Of the 170 open items, 17 are site specific, 110 involve information in the scope of an OL or FDA and/or Design Certification application, and 43 had not been resolved by the staff when it closed its review in October 1989. (Westinghouse submittals on many of these 43 open items, including its proposed resolution of Generic Safety Issues, Unresolved Safety Issues, post-TMI regulatory requirements, and outstanding PRA issues are yet to be reviewed by the staff.) In view of these open items and our concerns regarding the SER and the many unresolved severe accident issues, we indicated to the staff that its conclusions on page 25-1 of the draft SER were stated too strongly. The staff agreed to revise this language.

The Committee is not of one mind regarding the issuance of a PDA for the RESAR SP/90. On the one hand, there is merit to the argument that Westinghouse's application for the RESAR SP/90 PDA was made in good faith in 1983 under a different set of regulations and that it is now appropriate to document the reviews that have taken place to date and issue the PDA for potential future use as a reference design for an individual plant CP application or as the starting point for an FDA and/or Design Certification application. Both Westinghouse and the staff advocate this approach; neither believes that it can devote further resources to this effort.

On the other hand, we view the RESAR SP/90 SER as a mixed bag of staff evaluations that were performed over the seven-year period since the application was filed. Some are current and well done; others are poorly done and/or were performed years ago and do not meet the standards that we believe should be applied to a current SER. A major contributor to this problem appears to be the staff's reliance on the July 1981 Standard Review Plan (SRP) (NUREG-0800) in performing this review. This SRP needs updating to reflect the current situation for the licensing of ALWRs.

Some examples of our concerns with the staff's SER are:

3.1.1 SER Chapter 7, Instrumentation and Controls, references a staff review that was performed in 1979 for the Westinghouse RESAR 414 design. The staff concluded that the computer based integrated reactor protection system design for RESAR SP/90 is acceptable for a PDA on the basis of the "similarity" of the RESAR 414 design to that proposed for RESAR SP/90. It is our view that the staff should have developed improved standards for the review of such systems during this 11-year period. We are particularly concerned about the verification and validation of the software employed with computer based reactor protection systems. It appears that there is a need to augment existing staff resources with expertise in the computer science area so that appropriate standards can be developed for the review of computer based reactor protection systems. All of the proposed evolutionary and passive ALWRs employ such systems.

3.1.2 For materials used in the fabrication of pressure boundary components, Westinghouse has committed to follow applicable codes, standards, and regulatory guides. Many of these are not representative of current industry practice for such materials. We learned that Westinghouse has developed internal specifications for pressure boundary materials that presumably do reflect current industry practice. These were not submitted for the staff's review.

3.1.3 The proposed design employs water displacer control rods and associated control rod drive mechanisms, which is a new feature for Westinghouse plants. The SER describes the function of and strategy for use of these control rods. The SER, however, does not discuss the pressure boundary integrity of these new control rod drive mechanisms or the potential for reactivity insertion accidents that could result from misoperation of these control rods. Although Westinghouse submitted information on these subjects, the staff has not completed its review of this information. In general, we believe that new features of this kind should be thoroughly reviewed at an early stage of review.

3.1.4 Our review, which represents only a sampling effort, revealed a number of factual errors and inconsistencies in the SER; the staff has agreed to correct these errors. We believe that a review of the draft SER by Westinghouse, which has not yet had access to this predecisional document, would reveal additional errors that should be corrected. We recommend that this be done.

3.2 Comments on the PDA Document

The PDA states that the preliminary design information contained in RESAR SP/90 "complies with the requirements of 10 CFR Part 52, Appendix O . . . and is acceptable for incorporation by reference in applications for individual construction permits . . . " The PDA does not describe how this preliminary design information would be used in a future FDA and/or Design Certification application.

We were told by OGC that this results from the fact that Westinghouse has not made an application under 10 CFR Part 52.

Given the quality of the SER for this PDA, we are concerned with the language of the PDA that requires the staff and ACRS to utilize and rely on the "approved preliminary design" in their reviews of any individual facility construction permit application " . . . unless significant information which substantially affects the determination set forth in this PDA, or other good cause, is present." OGC advised us that this requirement would apply only to the staff and ACRS reviews of a CP application and that both entities would be able to revisit any issue in their review of any type of application that would lead to an OL. This is satisfactory to us but could present problems for the staff in dealing with a contested CP application.

4.0 Comments on the SP/90 Design

We have two concerns regarding SP/90 design features:

4.1 Our review of the NPB layout indicates that Westinghouse has provided many desirable features from the standpoint of separation of equipment trains for protection against fires and industrial sabotage. However, we are concerned about the location of the emergency diesel generators (EDGs) on the same floor and corridor from the control room. We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators.

4.2 The proposed RESAR SP/90 design employs a spherical containment. To deal with core/concrete interaction, the layout of the containment employs a cavity floor area beneath the reactor vessel that is based on the EPRI requirement of 0.02 m^2 per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all of the ALWRs.

5.0 ACRS Recommendations on the Issuance of a PDA

We believe, subject to the above comments, that the proposed design of the RESAR SP/90 NPB can be successfully completed and used in an application for an individual plant CP. Accordingly, we recommend that a PDA be issued for the proposed Westinghouse RESAR SP/90 NPB.

6.0 Concluding Remarks

Finally, we wish to commend the Westinghouse Electric Corporation, the Japanese APWR program participants, the EPRI ALWR Utility Steering Committee, and the EPRI staff for the effort they have expended in the development of this evolutionary design. The RESAR SP/90 design represents an important step forward in providing improved LWR designs that incorporate many of the lessons related to safety, performance, and reliability that have been learned by the nuclear power industry over the past 30 years.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft NUREG-1413, "Safety Evaluation Report Related to the Preliminary Design of the Standard Nuclear Steam Supply Reference System, RESAR SP/90" (Predecisional)
2. Draft Westinghouse Electric Corporation, Docket No. 50-601, Reference Safety Analysis Report (RESAR SP/90 Nuclear Power Block Standard Design), Preliminary Design Approval (PDA) (Predecisional) (Discussed during the November 8-10, 1990 ACRS full Committee meeting)
3. Letter NS-EPR-2675 dated November 1, 1982 from E. P. Rahe, Jr., Westinghouse Electric Corporation, to F. Miraglia, U.S. Nuclear Regulatory Commission, Subject: Westinghouse Advanced Pressurized Water Reactor Licensing Control Document

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

September 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: YANKEE ROWE REACTOR PRESSURE VESSEL INTEGRITY

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we discussed the degree and consequences of the Yankee Rowe reactor pressure vessel embrittlement due to neutron irradiation. Our Subcommittee on Materials and Metallurgy discussed this matter with representatives of the NRC staff and the Yankee Atomic Electric Company during a meeting on September 5, 1990. We also had the benefit of the documents referenced.

It has recently come to the staff's attention that the reference temperature nil ductility transition (RT_{NDT}) of parts of the Yankee Rowe pressure vessel may substantially exceed the temperature limits for action delineated in the pressurized thermal shock (PTS) rule (10 CFR 50.61). The main reason is that the Yankee Rowe core inlet temperature is about 50°F lower than that of other plants. Another reason is the higher nickel content of the lower vessel plate. These increase the rate of rise in RT_{NDT} with fast neutron irradiation.

The exact value of RT_{NDT} for the vessel is uncertain because of:

- Uncertainty in the copper and nickel content of the circumferential weld near the reactor vessel beltline.
- The absence of surveillance data for areas that appear to have the largest shift in RT_{NDT}, namely the circumferential weld and the lower plate of the vessel.

Assurance of vessel integrity is further hindered by:

- The absence of any inservice inspection for flaws in the reactor vessel beltline region. Such inspection has been infeasible due to the design of the vessel internals.
- Relatively low toughness (low upper shelf energy) of the plate and welds near the core.

Analysis of the various safety issues involved leads to the conclusion that PTS is the issue of most concern. One bright spot in this picture is that several features of the plant's design make it less susceptible to overcooling events than more modern plants.

The licensee and the staff have both arrived at estimates of the shift in RT_{NDT}. Both agree that the circumferential weld and the lower plate of the pressure vessel have the highest RT_{NDT}. However, in each case their estimates differ by about 150°F. The licensee's representatives argue that due to the particular microstructure of the steel in the vessel, the shift in RT_{NDT} is independent of irradiation temperature and nickel content. We do not believe these arguments are valid, and agree with the staff that temperature and nickel effects must be included in a valid estimate of the shift in RT_{NDT}. An additional difference between the staff and the licensee concerns estimates of the copper content of the circumferential weld. There being no measurements for the composition of the circumferential weld and a large spread in copper values found in other plants, the staff prefers to choose a bounding value. The applicant chose more of an average value. In view of the uncertainty in the value for the Yankee Rowe vessel, we would choose the staff's bounding value.

Given that RT_{NDT} values for parts of the vessel probably exceed those requiring action under the PTS rule, is there significant risk in operating the plant? The low probability of a PTS challenge leads to a low risk, even with a high RT_{NDT}. Thus, we agree with the staff that operation for one more cycle is acceptable, provided the licensee initiate an active program to better characterize the material in the vessel near the reactor vessel beltline. To do this the staff requires determination of the composition of the circumferential weld metal in the beltline by removing samples from the weld and development of an inspection method for the beltline welds and plate to depths of an inch below the inside surface of the vessel. Both of these have been required by the staff for completion before the startup of the 22nd fuel cycle (now scheduled to begin in early 1992). It is not clear that both can be achieved in that time, but certainly they should be accomplished in two fuel cycles.

The staff also requires "tests on typical Yankee Rowe base metal" to determine the effect of irradiation, austenitizing temperature and nickel content on embrittlement. It is doubtful that any tests that the licensee could perform during the next fuel cycle would convince us that the effects of temperature and nickel on embrittlement are substantially different from those established by the much more extensive studies already available. The effects are not well understood, and we believe prudence dictates tending more toward bounding values rather than best estimates based on limited new data that may become available.

However, the above will not adequately address the long-term operation of the plant. This is the lead PWR plant in the industry's Plant Life Extension (PLEX) program, and long-term operation with such large uncertainties in vessel integrity is unacceptable. The extended operation of this plant would be acceptable only if:

- A state-of-the-art ultrasonic inspection can be done on essentially all of the radiation affected inner surface of reactor pressure vessel, e.g., one that complies with Appendices VII and VIII of Section XI of the ASME Code. This inspection should also check for significant thinning in the lower head as a result of loose parts (irradiation capsules). Continued operation would be dependent on the absence of significant flaws.

- A reanalysis of the PTS question is made using well established compositions for the material in the beltline region, or using limiting values of copper and nickel. This analysis should also include the fact that the crack arresting ability of such material will be lower than more modern steel because of its low upper shelf energy. Such an analysis must show acceptable risk.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>References</u>:

1. Letter dated July 5, 1990 from John D. Haseltine, Yankee Atomic Electric Company, to Richard Wessman, NRR, transmitting Reactor Pressure Vessel Evaluation, dated July 9, 1990

2. Letter dated August 31, 1990 from Thomas E. Murley, NRR, to Andrew C. Kadak, Yankee Atomic Electric Company, Subject: Yankee Rowe Reactor Vessel, with Enclosure

2

Part 2: ACRS Reports on Generic Subjects

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

April 24, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SEVERE ACCIDENT RESEARCH PROGRAM

During the 360th meeting of the Advisory Committee on Reactor Safeguards, April 5-7 and April 18-19, 1990, we reviewed the Severe Accident Research Program (SARP) of the NRC. Our Severe Accidents Subcommittee discussed this program with the staff during a meeting held on March 20-21, 1990. We also had the benefit of the documents referenced.

During this review, emphasis was given to what the staff describes as its short-term program. Basically, the short-term program focuses on issues associated with early containment failure, e.g., BWR Mark I liner attack and direct containment heating (DCH). However, a description of the long-term program was also presented and discussed briefly. In what follows, we give a brief description of most of the elements of the SARP program, together with our comments and recommendations.

Adding Water to a Degraded Core

This investigation proposes to address a number of questions arising in connection with the in-vessel progression of severe core damage. The planned studies are said to address issues of in-vessel vapor explosion, thermal shock, and recriticality. Each study is an analytical investigation using a number of existing codes.

We are not convinced that the codes to be used are capable of providing the information being sought with sufficient validity that it can be used for the purposes listed, namely, the removal of uncertainty and the provision of information for use in the individual plant examination (IPE) program. Nor are we convinced that models that are to be developed can be demonstrated to be valid in a reasonable time, if ever. This program should receive the sort of analysis that is being developed in the Severe Accident Scaling Methodology (SASM) program discussed below.

Core Melt Progression

This is a program planned as a collaborative effort with the Federal Republic of Germany (FRG). Its purpose is said to be learning more about core melt progression in BWRs in order, presumably, to construct codes to describe the in-vessel melt progression process in this type of core. We were unable to determine how much additional information is needed in the regulatory process or whether the planned program will produce it in time for it to be used in, for example, the IPE program, which is apparently its earliest planned application.

This program should be examined further, using the techniques developed in the SASM program. We commend especially the lessons learned from that program and the "Evaluation Questions for Proposed Severe Accident Experimental Programs" as were discussed with our Subcommittee on March 20-21, 1990.

Examination of TMI-2 Lower Head and Lower Head Failure Analysis Plan

The sampling of TMI-2 lower head material described to us seems to be worthwhile, since we hope and believe that incidents that produce such an opportunity will be infrequent, and one should learn as much as feasible from this accident, especially if it will help prevent future accidents. However, we were disappointed at the response when we asked how the information being collected would be used by the NRC. We were told that it would be used to calculate "the margin to failure." However, when we asked what one would do with this margin there seemed to be considerable uncertainty.

The lower head failure analysis plan is a rather extensive and ambitious effort to model various postulated modes for bottom head failure. Effort will be made to use the information collected from the TMI-2 lower head examination to validate some of the models. In particular, we assume that effort will be made to understand why the relocation of a portion of the hot core into the lower head caused such a limited temperature rise in the body of the vessel head.

Presumably, some estimates of likely lower head failure timing can be made from the results, and since the timing of lower head failure has an influence on the time at which containment failure is likely to occur, this information may be useful in risk estimates. We did not, however, receive any information that would lead us to believe that significant thought has been given to how the information developed will be used in or useful to the regulatory process. We observe also that the number of variables likely to enter into a determination of lower head failure is so large and largely unpredictable that predictions of the likelihood of the various possibilities may be subject to very large uncertainties.

Severe Accident Scaling Methodology

Much experimental research is performed under conditions of size, geometrical configuration, temperature, pressure, and in some cases with materials that are different than what is expected in severe accident conditions in a large power reactor. It is usually hypothesized that if a computer code which has been constructed to model the progression of a severe accident can predict the results of properly constructed experiments, the code can, with appropriate allowances for the differences in the experimental conditions and what is expected in the reactor accident, predict at least the important features of a severe accident. These allowances, or scaling factors, require a careful analysis of both the experiments and the hypothesized accident. The process has not received the attention it should have had in much of the severe accident research performed for the NRC.

Members of the staff described a program which appears to be well designed and which, if it continues and if the results are applied, is likely to place future experimental work and code development on a much sounder basis. It will also provide guidance for any further experimental work that may be required, guidance that has frequently been unavailable in the past. We were impressed that the project manager was able to assemble an outstanding group of experts with representation from industry, the national laboratories, and academia, and was able to obtain significant cooperative effort from the group in performing the study. To ensure its applicability, the method of analysis that has been developed is currently being applied to the investigation of a DCH sequence. It gives promise of providing needed insight into this thorny problem, as well as providing guidance in the planning of other research programs.

The Probability of Liner Failure in a Mark I Containment NUREG/CR-5423

The authors of this report have collected, from a variety of sources, and have correlated a considerable body of information about the important phenomena that contribute to the processes that begin with severe damage of the reactor core and result in a pool of molten corium in contact with the metal structure that forms the boundary of the dry well of a Mark I containment. This, and their efforts to construct from the information a coherent picture of the core melt-vessel breach-attack of the liner sequence represent a significant contribution.

However, it is important that it be recognized for what it is, and for what the authors say it is, namely an attempt to take the existing information, to fill in the gaps of information needed to reach a conclusion about liner failure, using mostly engineering judgment, and to thereby construct a framework. This framework, given the existing information and the assumptions made to fill the gaps, permits the authors to reach conclusions about liner failure. And although the authors claim confidence in the conclusions they reach, with a few caveats, it would be unfortunate to use the results in making regulatory decisions without recognizing that conclusions about many of the important phenomena, for example the rate of release and the state of the corium from vessel breach, that have a significant impact on the final result, are supported primarily by the authors' judgment.

An important part of the report, and a part which was not available to us for our review (it is not yet complete), is Appendix F, that will contain the results of a peer review of the report. We have not had the rest of the report long enough to perform a thorough review. However, we do make the following observations:

- The approach used appears to be sufficiently similar to that developed in NUREG-1150 that the authors might have estimated the uncertainty in the results of their calculations. This would have added to the value of the report. During the presentation made to us, one of the authors argued that because computational uncertainty is not the only uncertainty in the result, it was not considered useful to estimate it. However the calculational process is the question at issue here, and unless some bounds can be set on the uncertainty of the results of the process, its value is diminished considerably.

- Even though the authors chose not to make a quantitative estimate of uncertainty, they are, having gone through this extensive study, in a unique position to identify, at least qualitatively, where the greatest contributors to uncertainties lie. They should be encouraged to do this, as well as asking others to identify them as they chose to do.

- Furthermore, the authors do not discuss whether the method used for those situations for which needed information is not available, i.e., estimating probability distribution functions, sampling these functions to get a range of possible values for the parameter of concern, and finally combining the results in a way which is something like calculating a mean, is any more nearly valid than estimating the mean value at each place where needed information is unavailable. The authors should be encouraged to justify that the method used is superior to simply estimating a mean value for uncertain parameters and using that value for further calculation.

Continuing Code Development

In our report of March 15, 1989 to then Chairman Zech, we noted that a review of NRC sponsored codes was being performed and that support for some codes that were found to be duplicative or no longer needed would be discontinued. This review has been completed and further support for several codes has been withdrawn. There has also been an increasing emphasis on documentation of those codes that exist as well as those being developed. We applaud this emphasis.

We were briefed on continuing development of two codes that are to be retained, CONTAIN and MELCOR, which are expected to provide much of the analytical capability which the staff will use in severe accident analysis. Unfortunately, the presentations and discussions were such that we were unable to obtain the information required for making any recommendation at this time. We will explore this further because the staff is expecting to make use of these codes in drawing conclusions about severe accident progression in both existing and new plants. MELCOR is, for example, to become the principal tool for calculating fission product sources.

Molten Core-Concrete Interactions

This experimental work is said to be needed because of continuing uncertainty about the contribution of Molten Core-Concrete Interaction (MCCI) to containment failure. The point was made that the contribution is primarily to late containment failure, and thus may be less important than contributors to early failure. However, because the staff expects that advanced reactor designers will assume that the debris produced during core melt will be coolable in the designs being proposed, the additional information being sought is deemed essential to advanced reactor review. It is also claimed that MCCI is an important part of the Mark I liner failure issue.

Integration of SARP with Foreign Research

In addition to their own research, the Division of Systems Research has a systematic program in place to learn from and, in some cases, to participate in the research of several foreign countries. This program seems effective.

Long-Term Research

To a considerable extent, the research proposed could be said to be more of the same. Most of what is described is justified on the basis that uncertainties need to be decreased in such areas as Modeling Severe Accidents, In-Vessel Core Melt Progression and Hydrogen Generation, Hydrogen Transport and Combustion, Fuel-Coolant Interactions, Molten Core-Concrete Interactions, Fission

Product Behavior and Transport, and Fundamental Data Needs. It will be recognized that these are not new, and indeed each has been an object of research almost from the beginning of the Severe Accident Research Program.

We do not have sufficient information to justify an endorsement of this program, although this may be because it is not yet well defined. We were unable to obtain satisfactory answers to questions such as:

- How much uncertainty is acceptable?
- How much will the proposed research reduce the uncertainty?
- Will the information obtained reduce risk, or will it merely permit less conservative approaches to design and operation of plants?

This program is another that should be subjected to the type of analysis suggested in the SASM program.

In connection with both the long-term and the short-term programs, we perceive a lack of communication between those planning the research and those who will use the results. It is indicative of the loose coupling between severe accident research and regulatory activities that in his summary of the research program, provided to the subcommittee, the Director of the Division of Systems Research commented that the "Agency doesn't have a definite regulatory use for severe accident data, i.e., no rule or regulation, no user needs letter." He did go on to indicate that there are a number of "indirect" uses. However, it appears to us that the main point he made is valid, and is a point of some concern. As early as 1975, WASH-1400 illuminated the risks associated with severe accidents. This led to the conclusion that absent a severe accident there is little or no risk to the public from the operation of nuclear power plants. Yet since that time, even in the light of the TMI-2 and Chernobyl accidents, little change in the regulations that govern the operation of nuclear power plants has occurred. Even for plants not yet licensed, there are virtually no new regulatory requirements dealing with the performance of the plant systems in the course of a severe accident. We have, for example, virtually the same rules governing containment performance requirements as we had in 1971. Of course it is required that new plant designs be accompanied by a PRA, but how the PRA is to be used in judging the acceptability of the design is undetermined. Under these circumstances, it is difficult to judge what new research in severe accidents is needed. Of course, it is possible that nothing more need be done, but aside from the Commission's Safety Goal Policy Statement and its Severe Accident Policy Statement, there has been no formal recognition of severe accidents, even for new plants.

It may be that the current emphasis on what happens in the plant after breach of the vessel is overdone. Examination of the results of most of the existing PRAs indicates that none show risks in excess of the Safety Goal quantitative objectives (not all of these, however, include seismic risk). However, several show core damage frequencies in excess of the sometimes proposed goal of 1E-4 per reactor-year. Thus, in a situation in which resources are limited, it may be that more emphasis should be placed on decreasing the likelihood of core damage. For example, for PWRs, many PRAs estimate that off-site risks are dominated by the ISLOCA (this is the case for Millstone-3, Seabrook, Surry, and Sequoyah, for example). Here phenomena occurring after vessel melt-through are of little consequence in risk determinations.

In the presentation to the subcommittee, the staff representatives stated that they believe uncertainties in the vessel failure scenario, and subsequent events, are the major contributors to risk uncertainties, based on PRA results. This is at least questionable in view of the risk attributed to seismic events and to human performance, and the large uncertainties associated with both of these.

Comments and Recommendations

There is much of deja vu in the proposed severe accident research. The same areas that were being explored at the beginning of the program almost ten years ago are still being investigated. The justification given by the Office of Nuclear Regulatory Research is that uncertainties exist which are large enough that regulation is difficult or impossible. However, there is little assurance that the proposed research will reduce the uncertainties to an acceptable value. Nor does there seem to be a very specific idea of what an acceptable value would be. This is probably not altogether the fault of the Office of Research.

A decision on what is acceptable is difficult to make, and requires, as a minimum, a close collaboration of the Office of Research with the Office of Nuclear Reactor Regulation. There appears to be an improvement in this collaboration, but from what we can tell, more teamwork on the issue of what research is needed is essential if the research is to be properly focused.

We are enthusiastic about the SASM program. Moreover, the approach that is being developed, if applied to planning the NRC's severe accident research program, can result in focusing the program to areas where it is most needed, and in making it more likely that the projects undertaken will produce useful information.

The MCCI work is a further pursuit of information on ex-vessel severe accident phenomena. Although we were not provided with enough information to reach firm conclusions concerning the worth of the proposed research, we observe that estimating the contribution of MCCI to late containment failure requires information beyond establishing the cavity area that will ensure quenching of core debris.

Concerning the programs discussed above we have the following recommendations:

- We recommend that the proposed research projects on Adding Water to a Degraded Core and Core Melt Progression not be undertaken until they are subjected to a review of the type developed in the SASM program. If they survive the review they will be much more likely to enhance the regulatory process.

- We were told that, in light of the staff's view of the success of the study described in NUREG/CR-5423, consideration is being given to applying this same type of analysis to the DCH issue. In our view, a SASM-type approach is likely to produce more useful information than will the NUREG/CR-5423-type analysis in its present state of development. We recommend that a SASM-type study be used as an alternative to the NUREG/CR-5423 approach.

- In connection with the TMI-2 vessel examination, we recommend that further thought be given to the way in which the information being collected might be used. We consider the examination worthwhile, but believe there must be applications beyond calculating the margin to failure.

  We recommend that the Lower Head Failure Analysis be subjected to the SASM process. If this study is to be done, it should have more of a relationship to regulatory needs than we are able to discern.

We recognize that this report may seem unduly critical. However, our comments reflect our perception that the various elements of the SARP lack focus. We do not attribute all of this lack of focus to the Office of Research. Part of it comes from the inability of the agency to deal with severe accidents in a regulatory context.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1365, "Revised Severe Accident Research Program Plan FY 1990-1992," August 1989

2. U.S. Nuclear Regulatory Commission, NUREG/CR-5423, "The Probability of Liner Failure in a Mark-I Containment," T. Theofanous, et al. (UCSB), February 1990

3. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Volumes 1 and 2, June 1989

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

May 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED GENERIC LETTER SUPPLEMENT ON INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES DUE TO EXTERNAL EVENTS

During the 361st meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 1990, we reviewed the NRC staff's proposals for Individual Plant Examinations for External Events (IPEEE) in furtherance of the Commission's policy statement on severe accidents. This matter was also discussed during a joint meeting of our Subcommittees on Extreme External Phenomena and Severe Accidents on March 27, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff, NUMARC, and the Yankee Atomic Electric Company. We also had the benefit of the document referenced.

The NRC staff proposes to issue a Supplement to Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, and a document in the form of a NUREG to provide guidance and recommendations regarding the scope and acceptable methodologies for the IPEEE. We have reviewed drafts of these documents and have no problems with what the staff is proposing. The guidance document is based on the report of the External Events Steering Group and describes both the staff's positions and the basis in terms of data, experience, and judgment for reaching those positions. The staff has worked closely with the licensed utilities, through NUMARC, in developing the basis for and the contents of the Generic Letter Supplement.

For seismic events, a seismic margin approach is an acceptable alternative to a seismic PRA. However, either method requires a plant walkdown by an experienced multidisciplinary team. We agree with this approach.

For internal fires, either a Level 1 fire PRA or a simplified fire PRA, with specified enhancements in either case, is permitted. A simplified fire risk evaluation method is being developed by

NUMARC, but has not yet been evaluated by the staff or by us. We find the proposed PRA approaches acceptable and expect to review the NUMARC method.

For other external events, such as high winds, flooding, and transportation accidents, a screening procedure not requiring a PRA is proposed. We find this acceptable.

The staff proposes to issue the Generic Letter Supplement after approval by the Commission. The final draft of the NUREG guidance document will not be issued until after a workshop is held to describe and discuss the recommendations. It seems likely that some changes may be required to the guidance document as a result of the workshop. Revisions to the Generic Letter Supplement also may be desirable. In either case, we would like to have the opportunity to review such changes and to provide our comments to you, as appropriate.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
Memorandum dated March 8, 1990 from W. Minners, Office of Nuclear Regulatory Research, NRC, for R. F. Fraley, ACRS, Subject: Proposed Generic Letter on Individual Plant Examination for Severe Accident Vulnerabilities Due to External Events (IPEEE), with enclosed Predecisional Draft Commission Paper, with its enclosures: (1) Generic Letter (Draft Predecisional) and (2) NUREG-XX IPEEE Guidance, Draft for Comment

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

June 13, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: DRAFT STUDY ON SOURCE TERM UPDATE AND DECOUPLING SITING FROM DESIGN

During the 362nd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 1990, we reviewed the NRC staff's Draft Study on Source Term Update and Decoupling Siting from Design. This matter was also discussed during our 361st meeting, May 10-11, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

At present, siting issues, including the definitions of the Exclusion Area (EA) and Low Population Zone (LPZ), are governed by 10 CFR Part 100, Reactor Site Criteria, which sets limits on the exposure of an exposed individual in the event of certain hypothetical accidents. The necessary calculations require assumptions about the amount of radioactivity released to the containment in those accidents, the so-called source term.

It is customary to use for the latter an old AEC report, Technical Information Document 14844, dated March 23, 1962. It has been recognized for about ten years that that report grossly overestimates radioactive releases in a typical accident, and misrepresents their forms. Consequently there has been in this period a leisurely effort to "update the source term."

The staff soon recognized that the effects due to possible reduction of the source term, and reduced probability of an accident, could combine with the requirements of 10 CFR Part 100 to make possible the licensing of plants with uncomfortably close boundaries, perhaps even in a metropolitan area. To avoid this, the staff proposed that the siting question be decoupled from the source term upgrade, so that the customary sizes of the EA and LPZ could be preserved, as encapsulated in Regulatory Guide 4.7, General Site Suitability Criteria for Nuclear Power Stations. This is a matter of preserving the answer, in the face of creeping safety improvements, by rephrasing the question.

In the end, the staff considered a number of options, including a revision of 10 CFR Part 100 through rulemaking, and concluded that they were all so difficult that one ought to proceed by first updating the source term to accommodate current technical understanding. Then the tentative proposed solution to the siting problem is to "encourage" conformance to Regulatory Guide 4.7, in effect substituting a regulatory guide for rulemaking.

We support (as we always have) the effort to adjust the source term to reflect current knowledge. Since it appeared at our meeting that the staff is not itself entirely clear about its position on siting, we cannot yet provide definitive advice on that aspect of the problem. Perhaps, since no one is now proposing other than remote siting of nuclear power plants in the United States, the question is moot.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>Reference</u>:
Draft Commission Paper from James M. Taylor, Executive Director for Operations, Subject: Staff Study on Source Term Update and Decoupling Siting from Design (Predecisional), transmitted by memorandum dated May 25, 1990 from Warren Minners, Office of Nuclear Regulatory Research, for Raymond F. Fraley, ACRS

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

November 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: REVIEW OF NUREG-1150, "SEVERE ACCIDENT RISKS: AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS"

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we discussed the second draft of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." The Committee had previously discussed this matter with the staff and its consultants and with Dr. Herbert Kouts, Chairman of the Special Committee to Review the Severe Accident Risk Report. Our Subcommittees on Severe Accidents and Probabilistic Risk Assessment discussed this report during a number of joint meetings with members of the staff, Sandia National Laboratories (SNL) and the American Nuclear Society (ANS) Special Committee (Dr. Leo LeSage, Chairman). We also had the benefit of the documents referenced.

1. <u>INTRODUCTION</u>

In this report, we first offer some general comments. We then offer recommendations concerning the publication of NUREG-1150 and provide comments and cautions concerning interpretation or use of some of the components of this document. And finally, we provide more detailed comments on some key parts.

We have reviewed the reports prepared by the ANS Special Committee and by the Special Committee to Review the Severe Accident Risk Report appointed by the Commission and found them helpful. We have no serious disagreements with either of these reviews, nor with their findings.

2. <u>GENERAL COMMENTS</u>

The work described in this draft of NUREG-1150 is an improvement over that described in the first version entitled, "Reactor Risk Reference Document." Many previously identified deficiencies in the expert elicitation process have been corrected. The exposition and organization of the report have been improved. The presentation of results is clearer. There is considerable information that was not in the original version.

The portion that deals with accident initiation and development up to the point at which core heat removal can no longer be assured is unique, compared to other contemporary PRAs, in that a method for estimating the uncertainty in the results has been developed and applied. This method and its application are significant contributions. Although the larger contributions to uncertainty in risk come from the later parts of the accident sequences, this portion is enhanced also by an extensive identification of events that can serve as accident initiators as well as an associated set of hypothesized event trees. This information should be of considerable assistance to licensees in the performance of an Individual Plant Examination (IPE). It should also be useful to plant operators and to designers.

The formulation of a more detailed representation of accident progression after severe core damage begins, and an improved description of containment performance, contribute some additional information to this important area. However, understanding of many of the physical phenomena that have an important bearing on this phase of accident progression is still very sparse, and the report may give the impression that more is known about this portion of the accident sequence than is actually the case.

The part of the sequence that begins with the release of radioactive material outside the containment is treated by a relatively new and unevaluated code system. Furthermore, there is no estimate of the uncertainties inherent in the calculations that describe this part of the sequence. Those who use the quantitative values of reported risk must recognize that these uncertainties are not accounted for in the calculated results.

3. RECOMMENDATIONS

We recommend that the current version of NUREG-1150, with the corrections suggested by several of those who have already reviewed it in detail, be published. However, its results should be used only by those who have a thorough understanding of its limitations. Some of these limitations are discussed in subsequent sections of our report.

Since the supporting documents upon which NUREG-1150 depends could be helpful to those who perform an IPE, we recommend that these also be published as soon as feasible.

Both the Commission and the ACRS have raised questions about generic conclusions that might result from a careful examination of the results of this study. It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost five years and seventeen million dollars to complete. We recommend that the Commission encourage the staff to mine more deeply the wealth of information that has been collected in the course of this study in an effort to identify generic conclusions that might be reached (see Section 5.5 of this letter).

4. COMMENTS AND CAUTIONS CONCERNING USES OF THE MATERIAL IN NUREG-1150

We discuss below certain areas in which the methods or results should be used with caution.

4.1 Differences Among Levels of the PRA

The phenomena which contribute to sequence progression in Level 1 are generally well understood. Power plant or other related experience with system and component performance has provided sufficient data to permit predictions of sequence progression with considerably greater confidence than for those parts of the sequence described in Levels 2 and 3. NUREG-1150 is unique in the amount of effort that went into estimating uncertainties in the calculated Level 1 results. It is our view that the results of Level 1 can be used with more confidence than those of Levels 2 and 3. However, as other reviewers have reported, there are recognized deficiencies in the state-of-the-art treatments of human performance; and this report is not free of those deficiencies. In addition, some possibly important initiators, e.g., those at low power operation or at shutdown, and sequences initiated by fire, are either treated superficially or are neglected altogether.

The Level 2 analyses in NUREG-1150 include more detailed containment event trees than those found in any previous PRA. However, we have some concern that the amount of detail may lead to a conclusion that much more is known about the phenomena in this area than is actually the case.

Since there is a dearth of information concerning many of the phenomena that determine severe accident progression, expert elicitation was used most extensively in the Level 2 portion of the PRAs. There is general agreement that the techniques used for eliciting expert opinion in preparation of the second draft were significantly better than those used for the first draft. However, with insufficient information there can be no experts. Thus, use of the term "expert opinion" in a description of some of the Level 2 work may be misleading. (Further comments about the expert elicitation process are given in Section 5.3). We applaud efforts to improve on the Level 2 treatment of previous PRAs. We nevertheless believe that the results from Level 2 presented in this latest draft must be regarded as having major uncertainties in both calculated mean values and in estimated uncertainties.

The MELCOR Accident Consequence Code System (MACCS) was used for the consequence calculations of Level 3. Use of MACCS is a departure from many existing PRAs that use the Calculation of Reactor Accident Consequences (CRAC) series of codes. MACCS is a relatively new code, still under development. It has been neither benchmarked nor validated. Thus, in addition to the uncertainties inherent in the physical phenomena that enter into consequence modeling, additional uncertainties are introduced by the use of a new and relatively untested code.

No effort was made to estimate the uncertainties in the Level 3 calculations. Thus, the estimates of uncertainties in risk that are given in the report are only those arising from the uncertainties calculated for Levels 1 and 2. It is our judgment that the uncertainties in modeling the consequences of a release can be at least as large as those estimated for Level 2. For example, the health effects, especially for low dose exposures, are subject to large uncertainty, and the exposures themselves depend on actions (e.g., evacuation, sheltering, interdiction of land and crops) for which the uncertainty in prediction is largely unknown.

4.2 Assumptions Made in Screening

Users of the report should be aware of the assumptions made in the screening process for low-probability, high-consequence events. For example, the analysts assumed that the probability of total loss of DC power was less than 1×10^{-7} per year and thus could be neglected. The same assumption was made for loss of all service water. Thus, those who use the results in IPE work should recognize that these assumptions may not be valid for all operating plants.

4.3 Credit for Decay Heat Removal by Feed and Bleed

The success of the feed and bleed operation is highly dependent on human performance. Everyone seems to agree that there are large uncertainties in its treatment in this report. In addition, it is likely that the performance of valves, which must function if this maneuver is to be successful, are not well represented by the data for valve performance used in the calculations.

4.4 Performance of Motor-Operated Valves

There is now a significant body of evidence which indicates that the failure probability used to describe the operation of certain key motor-operated valves is too low. This may have an important bearing on the outcome of several accident sequences described in the report.

4.5 Contribution of Pump-Seal Failure to the Risk of Small Break LOCAs

We believe that more recent information and some new seal designs developed since the study was made would lead to a prediction of risk less than that reported.

4.6 Containment Performance

The lack of information about many of the physical phenomena that determine the performance of a containment system in a severe accident situation is such that only educated guesses can be made for some sequences that might make significant contributions to risk. Although the large number of event trees developed in the containment analyses is indicative of what was hypothesized by the analysts, the amount and quality of information concerning a number of key phenomena that determine behavior at branch points are low. The difficulty of arriving at a result with significant confidence is illustrated by two examples. In the analysis of the performance of the Mark I containment used in early BWRs, the experts in the original study predicted a large conditional probability of early failure. In the second study a different group of experts produced a bimodal distribution because part of the panel concluded that the probability of early failure was high, and part considered it low. A second example is the calculation of risk produced by postulated direct containment heating (DCH). In the first study, the calculated risk due to DCH for PWRs with large dry containments was a major contributor to the total risk. In the second version, its contribution was significantly less. In neither case had there been a major change in the information about relevant physical phenomena available at the time of the first study. Further, we find no consideration of the impact of ex-vessel steam explosions on early containment failure. There is little unambiguous guidance here for a licensee performing an IPE.

5. AREAS FOR SPECIAL COMMENT

In this section, we provide more detailed comments on some areas that appear to us to deserve special attention.

5.1 Fire Risk

The fire contribution to core-damage probability was estimated for two plants using insights gained during previous fire PRAs and studies, the latest methods and data bases developed under NRC sponsorship, and the benefits of extensive plant walkdowns. The methods and data used were probably the best available at the time the reported work was performed. Nevertheless we conclude, on the basis of later information, that the results should be viewed as being incomplete. The models used were not able to take full account of several issues identified by SNL in a scoping study of fire risks that was completed more recently. These are issues that have not been adequately considered in past fire risk studies and may increase the risk. Of particular concern are seismic-fire interactions, adequacy of fire barriers, equipment survival in the environment generated by the fire, and control systems interactions. The PRA for the LaSalle nuclear plant, which is nearing completion, may provide insights concerning the risk importance of these issues.

5.2 Seismic Risk

The seismic PRAs for the Surry and Peach Bottom nuclear plants were performed using two quite different representations of the seismic hazards. The results however, at least for sequences leading to core damage, were similar in terms of which accident initiators and sequences were important. This tends to support the acceptability of using the seismic margin approach rather than a PRA in the search for plant-specific seismic vulnerabilities in the IPE-External Events (IPEEE) program. However, the success of either approach in finding vulnerabilities depends strongly on walkdowns to identify those systems and components to be evaluated. Knowledge of what to look for is derived chiefly from PRAs done on other plants, and these have tended to focus primarily on core damage rather than releases of radioactive material to the environment. Although containments are usually quite rugged seismically, this is not necessarily true for containment cooling systems, containment isolation systems, etc.

Although the two seismic PRAs in NUREG-1150 have been carried through Level 3, these results have not been reported. We believe that these results might provide valuable insights about seismic vulnerabilities of containment systems.

5.3 The Expert Elicitation Process

There is general agreement that the use of expert elicitation in the preparation of the results in this draft of the report is improved compared to that used for the first version. However, we have reservations about some parts of the application of the process. For example, during our discussions of the choice of the participating experts we got the impression that an effort was made to choose participants in such a way that a wide spectrum of viewpoints would be represented. This was defended as proper, based on the assumption that unless this wide spectrum of opinion was represented, the uncertainty in expert opinion would not be appropriately accounted for. We found this argument unconvincing, and would have preferred to see individuals chosen primarily on the basis of their knowledge and understanding of the phenomena being considered. Furthermore, we were told that the budget for the study provided only enough funding to support the participation of about 20 percent of the experts who served on the panels. The remainder were drawn from the NRC staff or from organizations with contractual relationships to the NRC. This biased the selection toward people whose organizations depend upon the NRC for support. We also observe that the membership of the panels seems to have been dominated by analysts in contrast to those who have done significant research on phenomena of importance to the accident sequences being described.

5.4 Source Term Description

The staff, or at least that part of it closely associated with this study, has discarded for future use the Source Term Code Package (STCP) that was one of the resources used by the expert panels in the preparation of NUREG-1150. The expert elicitation method is too resource intensive to be used generally. At this time, only the MELCOR code is available to the staff for source term calculation. Although it appears to be an improvement over the STCP, it is not yet fully developed, nor is it generally available in its current form. Some method for calculating a source term will be needed by the staff and its contractors for performing or reviewing PRAs, as well as for other tasks, such as a revision of the siting rule.

5.5 Lack of General Conclusions

We have asked the staff whether the results reported in NUREG-1150 shed any light on the risk expected due to operation of the population of plants now licensed. With few exceptions, it is the staff's view that one can tell little or nothing about the expected risk of plants not studied from the results of the study of these five plants in NUREG-1150. In spite of these statements, however, those who prepared the report propose that applications will include evaluation and resolution of generic issues and prioritization of future research and prioritization of inspection activities. If, as we were told, the results from the analyses of these plants have little or no generic significance, application of these results must be made with considerable caution.

We believe that the large amount of information collected as input to the calculations made during this study, and the results of the large number of analyses undertaken, must surely permit some more general conclusions to be drawn than we find in this report. For example, the risk calculated for each of the five plants analyzed (although calculated only for internal initiators) falls within the Quantitative Health Objectives (QHOs) set forth in the Safety Goal Policy Statement. Each was designed and constructed and is operating within the rules and regulations promulgated by the Commission. There must be some significance in the fact that plants supplied by a number of different vendors, constructed at different locations, under supervision of different organizations, over a period of more than a decade, with rather different balance of plant configurations, and different containments, nevertheless fall within the QHOs. Is application of the NRC's regulations achieving the objectives of the NRC Safety Goal Policy?

Another area of interest is the risk reduction achieved by some recently promulgated rules. The report indicates that station blackout is a significant risk contributor for three of the plants studied. Answers to questions we asked during our meetings with the staff indicated that some of the plants analyzed had implemented most of the requirements of the Station Blackout Rule, while others had only just begun the process. Could one draw any conclusions from the plants studied as to the risk reduction to be expected from implementation of the Station Blackout Rule? Or could one estimate the risk reduction for some "average" plant? This would be interesting, since in the typical cost benefit analysis associated with backfit it is assumed that some such conclusion can be drawn about plants generally. It would be useful to see what an examination of these five plants would indicate.

The five nuclear power plants chosen for the study were selected partly on the basis of the different types of containment represented. We find little or no discussion of relative containment performance or identification of containment designs that might be expected to have superior mitigation capabilities. For example, in light of the containment being proposed for the Advanced Boiling Water Reactor (ABWR), it would be helpful to have any information or conclusions that were developed during the course of the study as to relative efficacy of the containment being proposed for that design as compared to the Mark I or the Mark III containments. Or, for large dry containments, does the subatmospheric operation of the Surry system provide a substantial decrease in risk (because, for example, of its continuous indication of leak tightness) as compared to a large dry containment operated at atmospheric pressure?

Although it may not be feasible to make major changes in containments of reactors now in operation, it is possible to choose containments with superior mitigation characteristics for nuclear plants not yet constructed. It might even be feasible, as a result of the study, to recommend a containment design that combines the best features of several of the existing systems. If in the course of this study information has been developed that could be used to reduce the conditional failure probability of containment, given severe core damage, the risk uncertainty in new designs might be reduced without requiring any additional studies of core damage progression.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Volumes 1 and 2 (Second Draft for Peer Review), dated June 1989.
2. American Nuclear Society, "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," L. LeSage (Chairman), dated August 1990.
3. U.S. Nuclear Regulatory Commission, NUREG-1420, "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," H. Kouts (Chairman), dated August 1990.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Reactor Risk Reference Document," Volumes 1, 2, and 3, Draft issued for comment, dated February 1987.

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

March 13, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM - PROPOSED RECOMMENDATIONS FOR MARK II, MARK III, ICE CONDENSER, AND DRY CONTAINMENTS

During the 359th meeting of the Advisory Committee on Reactor Safeguards, March 8-10, 1990, we discussed the staff's proposed recommendations from the Containment Performance Improvement (CPI) program for plants with Mark II, Mark III, ice condenser, and dry containments. The staff intends to inform licensees with such plants of these recommendations in a supplement to Generic Letter 88-20 (Reference 1) and, by this action, will consider the CPI program completed. Our Containment Systems Subcommittee discussed this matter with the staff during a meeting on February 6, 1990. We also had the benefit of the documents referenced.

The CPI program is one element described in SECY-88-147, "Integrated Plan for Closure of Severe Accident Issues." Other elements in this plan are the Individual Plant Examination (IPE) program (Generic Letter 88-20), severe accident research, external event resolution, accident management, and improved plant operation. The CPI program was to identify any severe accident vulnerabilities that appeared to be generic to plants with a given type of containment. It was then to develop new regulatory requirements or guidance for reducing those vulnerabilities. Recommendations were to be derived by the staff and its contractors through study of risk analyses reported in NUREG-1150, other PRAs, and results from severe accident research. The intent was to identify any new requirements in the near term so that licensees could implement them along with any plant improvements identified in their own IPE efforts.

Mark I containments for BWRs were considered first. Staff guidance for Mark I plants was provided in Supplement No. 1 to Generic Letter 88-20 and in Generic Letter 89-16. We provided comments on the Mark I CPI program in our report dated January 19, 1989 to then NRC Chairman Zech.

The remaining four containment types have been considered as a group. The staff reports that it has "found no improvements for these containment types that would warrant generic implementation for all containments of a given type." However, it has identified some ways, unique to each containment design, in which plants may be particularly vulnerable to severe accident threats. While the staff has decided not to prescribe remedies for the generic problems it has identified, it does intend to provide licensees with technical insights and information that the staff believes to be of particular import. This will permit these lessons to be factored into IPEs and accident management programs that are being initiated by licensees. Summaries of the staff's concerns for each containment type are given in the proposed supplement to Generic Letter 88-20. More technical details will be provided in a series of reports that are being prepared by contractors to the staff and are expected to be available during June 1990.

The approach proposed by the staff is appropriate and we endorse the proposed supplement. We agree that the CPI program can now be terminated. As stated in our report of January 19, 1989 on the Mark I CPI program, the IPE program can be an effective and efficient means to identify and ameliorate risk-significant issues related to containment performance. The IPE and accident management programs will benefit by considering conclusions from these staff studies.

However, we recommend that the staff caution the licensees not to focus exclusively on the set of issues raised by the CPI program. For one thing, the designs analyzed in NUREG-1150 do not adequately represent the full spectrum of plants. For another, conclusions about risk and phenomena are subject to large uncertainties. Licensees should retain a broad perspective in their studies. The original intent of the IPE program, that is, to search "for possible ... 'outliers' that might be missed absent a systematic search," is applicable to issues of both prevention and mitigation.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:
1. Memorandum dated February 22, 1990 from Warren Minners, Director, Division of Safety Issue Resolution, RES, to Raymond F. Fraley, ACRS, Subject: ACRS Review of Supplement 2 [sic] to Generic Letter 88-20, Individual Plant Examinations, with enclosures:
   (a) Proposed Draft Supplement to Generic Letter 88-20, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities" (Predecisional)
   (b) Draft memorandum for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Recommendations of Containment Performance Improvement Program for Plants with Mark II, Mark III, Ice Condenser, and Dry Containments (Predecisional)
2. Letter dated November 23, 1988 from D. Crutchfield, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f) (Generic Letter 88-20)
3. Letter dated September 1, 1989 from James G. Partlow, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Installation of a Hardened Wetwell Vent (Generic Letter 89-16)
4. Letter dated August 29, 1989 from James G. Partlow, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f) - Generic Letter 88-20, Supplement No. 1
5. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants (Second Draft for Peer Review)," Volumes 1 and 2, June 1989
6. SECY-88-147, Memorandum dated May 28, 1988 for the Commissioners from Victor Stello, Executive Director for Operations, Subject: Integration Plan for Closure of Severe Accident Issues

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

January 18, 1990

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

Dear Mr. Taylor:

SUBJECT: RESOLUTION OF THE INTERFACING SYSTEMS LOCA ISSUE

During the 357th meeting of the Advisory Committee on Reactor Safeguards, January 11-12, 1990, we discussed the subject topic with members of the NRC staff. This issue was also discussed during our 356th meeting, December 14-15, 1989. Our Subcommittee on Thermal Hydraulic Phenomena considered this issue during its meeting on December 7, 1989.

The interfacing systems loss of coolant accident (ISLOCA) has been identified by the NRC staff as a problem of sufficient risk potential that a special program for its resolution is warranted. Such an event creates the potential for loss of two of the three barriers to fission product release, and if it occurs, is likely to lead to early fission product release outside of containment. Although earlier studies, including those reported in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," indicate that such an event has a low probability, the staff members who have undertaken the special program do not believe the previous studies have accurately represented human error contributions to the likelihood that such an event occurs.

The NRC ISLOCA program includes evaluation of available PRA analyses of accident sequences that may lead to such an event. Special emphasis is to be given to the human reliability contribution to initiation of such a sequence, and to ways in which its consequences may be mitigated. Engineering analysis of the low pressure piping systems will be carried out to determine where leaks or breaks could occur. A program of selected PWR plant audits is also under way. The results of these studies will be of value to the IPE effort in general, as well as to the ISLOCA issue, and the studies are encouraged. Special attention should be given to the environmental effects and flow-induced mechanical impact on equipment in the vicinity of the leak if the results are to be meaningful. Efforts should be made to ensure that the study results are broad enough to be applicable to BWRs. Our concern lies in how the results of these studies will be used.

Information provided by the staff leads us to conclude that causes of and optimal mitigation strategies for ISLOCA events are likely to be highly plant specific. In addition, important ISLOCA sequences apparently involve complex human actions that are not well modeled even in state-of-the-art PRAs. While the plant-specific nature of ISLOCA would seem to make it a logical candidate for the IPE process, the staff expressed concern that the PRAs that are likely to be used by licensees in performing their IPEs will not adequately deal with ISLOCA. Three approaches to resolving this issue were discussed:

(1) Information developed by the staff in its ISLOCA program could be used to modify PRAs used in IPEs so that ISLOCA is adequately analyzed. This is probably not practical and could undesirably delay completion of IPE programs.

(2) Information developed by the staff in its ISLOCA program could be used to develop a resolution and set of licensee requirements entirely separate from the IPE program. We believe this would tend to unnecessarily burden licensees with demands on their engineering and other resources and interfere with efforts to efficiently manage their IPE programs. We would not favor this option unless the staff program indicates ISLOCA might be an unexpectedly high contributor to plant risk.

(3) Information developed by the staff in its ISLOCA program might be furnished to licensees for incorporation into their IPE programs without the expectation that it would be comprehensively included in PRAs. We believe that PRAs should be regarded only as one, albeit important, tool and source of information to be used by licensees in their IPE programs. As a general premise, information from the ISLOCA program, resolution of GSIs and USIs, and many other sources can and should be used in IPEs, whether or not formally included in PRAs.

We recommend option number three as making the most efficient and effective use of staff and licensee resources.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

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

October 11, 1990

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

Dear Mr. Taylor:

SUBJECT: NRC COMPUTER CODES AND THEIR DOCUMENTATION

During the 366th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 1990, we continued our deliberations on the subject of the development of NRC's computer codes and their associated documentation. This topic was previously discussed during our 365th meeting, September 6-7, 1990. It was also discussed during a joint meeting of the Decay Heat Removal Systems and Thermal Hydraulic Phenomena Subcommittees held on August 28, 1990, in Idaho Falls, Idaho.

A portion of the regulatory process depends heavily on the results of calculations done for the NRC by the national laboratories or other contractors. The codes used for these calculations range from thermal hydraulic codes like RELAP5 or TRAC to severe accident codes like SCDAP or MELCOR. Many of these codes are poorly documented, thus leaving one unable to determine either their capabilities, or perhaps more importantly, their limitations. In some cases, it appears that even the cognizant NRC staff representatives are not sufficiently knowledgeable of a given code's content.

The NRC has a responsibility to make the basis for its computer codes as scrutable as it requires of the industry. Many code developers consider the documentation phase of the code development process distasteful. Nevertheless, the RES program managers should see that adequate documentation is provided, particularly for models and correlations and for developmental assessment. We have seen evidence that they have not done so. One of the central problems is the tendency to defer the preparation of such documentation until the end of the program. Although such a deferral may be understandable, given the natural progression of the development program, it is essential that program management ensures that documentation is provided in a timely manner and within budget.

The August 28, 1990 Subcommittee meeting was held to review the nearly completed work related to the development of the RELAP5/MOD3 thermal hydraulic code. Discussions during this meeting provided evidence that the associated documentation was incomplete. The contractor personnel were new to the program and not well enough acquainted with the code's details to respond to questions from the Subcommittee. The potential exists for similar problems with the completion of the development program for the TRAC-PF1/MOD2 code. Deliberate attention by RES program managers is needed to ensure the documentation for these codes is adequate.

Another example that illustrates our concern involves the thermal hydraulic code known as REMIX, which has been used by the NRC to evaluate the potential for pressurized thermal shock given certain accident scenarios. Relevant experimental data were generated as part of the cooperative 2D/3D program, among the United States, Germany, and Japan, and these data were compared with REMIX code calculations. Although a Research Information Letter citing this work was issued in 1988, a report documenting these comparisons has never been issued by the NRC. Recent review of the Yankee Rowe pressurized thermal shock issue would have been well served by knowing how well the downcomer fluid temperature can be predicted, using a code such as REMIX, at the beltline welds following a small break loss of coolant accident.

Many millions of dollars have been spent on the development of the computer codes used by the NRC, nearly $20 million for RELAP5 alone. The NRC should make sufficient funding and resources available to ensure that the documentation associated with the development of the agency's codes is adequate.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>Reference</u>:
Memorandum dated August 24, 1988, from Eric S. Beckjord, Office of Nuclear Regulatory Research, for Thomas E. Murley, Office of Nuclear Reactor Regulation, Subject: "Research Information Letter No. 155, Full Scale Fluid Mixing Test Results in Support of Pressurized Thermal Shock Resolution."

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

June 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: GENERIC ISSUE-84, COMBUSTION ENGINEERING PLANTS WITHOUT POWER OPERATED RELIEF VALVES

During the 362nd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 1990, we reviewed the staff's proposed resolution of Generic Issue-84, "CE Plants Without PORVs." We had the benefit of discussions with members of the NRC staff and representatives of Combustion Engineering Incorporated (CE). We also had the benefit of the referenced document.

Generic Issue-84 (GI-84) applies to six operating CE units that do not have PORVs or other means to rapidly reduce reactor coolant pressure through the venting of steam from the system pressurizer. These are the six CE units designed after 1970: San Onofre Units 2 and 3; Waterford Unit 3; and Palo Verde Units 1, 2, and 3. These units do have enhanced capability to reduce pressure by means of essentially safety grade auxiliary pressurizer spray systems.

This generic issue was established to determine if the capacity for "feed and bleed cooling" afforded by PORVs should be required for these units. Most pressurized water reactors can "bleed" through PORVs and "feed" with high pressure makeup pumps as an emergency means for removing decay heat from the core. While this cooling mode is believed to be useful in reducing risk of core overheating in some circumstances, the NRC has not made feed and bleed capability a requirement.

In its report dated December 15, 1981, the Committee expressed concern over a lack of means to feed and bleed in these plants during its review of the CE System-80 standard plant design. At the Committee's request, studies of the pros and cons of installation of PORVs on the CE plants were conducted by NRC and CE. These studies indicated ambiguity as to whether there would be a small reduction or a small increase in risk resulting from a backfit addition of PORVs. In 1983, the Committee agreed with a staff decision to incorporate this issue into the then ongoing effort on

Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." However, the ultimate resolution of USI A-45 in 1988 did not explicitly address the PORV concern, which was therefore carried on as GI-84.

The staff has now proposed that installation of PORVs not be required for the affected plants. The reasons cited for this decision are that such action is not required to meet any of the Commission's regulations, and that a cost benefit analysis, as called for under provisions of the backfit rule, indicates such system modifications are not justified. In general, risk analyses, which are an important input to the assessment of cost and benefits, lack the accuracy needed to make decisions about the very small differences in risk, plus or minus, that could be created by addition of PORVs to these plants. We also note that the risk analyses that have been conducted for resolution of GI-84 were limited in that they did not include consideration of external events as initiators. Nevertheless, we believe that these analyses have been useful and we concur with the staff recommendations.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
Draft Commission Paper, "Evaluation of the Need for Primary System High Capacity Manual Venting Capability on Combustion Engineering (CE) Plants Without PORVs (GI-84)," transmitted by memorandum dated April 27, 1990 from Warren Minners, Office of Nuclear Regulatory Research, NRC, to Raymond F. Fraley, ACRS.

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

August 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE B-56, "DIESEL GENERATOR RELIABILITY"

During the 364th meeting of the Advisory Committee on Reactor Safeguards, August 9-11, 1990, we reviewed the NRC staff's proposed resolution of Generic Safety Issue (GSI) B-56, "Diesel Generator Reliability." Our Subcommittee on AC/DC Power Systems Reliability also reviewed this matter during a meeting on August 8, 1990. During these reviews, we had the benefit of discussions with representatives of the NRC staff and of NUMARC. We also had the benefit of the documents referenced.

In our view, this proposed resolution includes unjustified imposition of maintenance requirements on the licensees, in contravention of the Commission's decision to defer issuance of a maintenance rule pending assessment of licensees' maintenance programs.

The proposed resolution of GSI B-56 involves two steps. First, Section C.6 of proposed R.G. 1.9, Rev. 3, contains an explicit example of a diesel generator reliability program, including maintenance, with detailed checkoff and corrective action lists. Second, the staff proposes to require adoption of R.G. 1.9, Rev. 3, by a generic letter pursuant to 10 CFR 50.54(f).

As background, GSI B-56 is related to the Station Blackout Rule (10 CFR 50.63). The staff issued R.G. 1.155, "Station Blackout," to provide guidance for compliance with this rule. R.G. 1.155 identified the need for a reliability program to achieve and maintain diesel generator minimum reliability levels of 0.95 or 0.975 per demand, depending on the blackout duration coping requirements calculated for a particular plant.

R.G. 1.9, Rev. 3, provides guidance for a reliability program by integrating into a single regulatory guide pertinent guidance now addressed in R.G. 1.9, Rev. 2, R.G. 1.108, Rev. 1, and Generic

Letter 84-15. In addition, R.G. 1.9, Rev. 3, endorses IEEE Standard 387-1984. This guide also describes a means for meeting the minimum diesel generator reliability goals contained in R.G. 1.155.

In developing the guidance contained in R.G. 1.9, Rev. 3, for the diesel generator reliability program, the staff has taken cognizance of related industry initiatives and programs, and for the most part is consistent with current industry practices. Both the staff and the industry seem to be in agreement concerning R.G. 1.9, Rev. 3, except for those parts of Section C.6 and accompanying figures and tables that prescribe in detail the requirements for a diesel generator reliability program.

NUMARC maintains that the licensees have committed to monitoring diesel generator reliability, and have docketed their commitments to maintain the chosen target reliability levels to comply with the Station Blackout Rule. NUMARC considers that these commitments together with their initiatives are sufficient to ensure acceptable diesel generator reliability.

Both the staff and NUMARC agree that diesel generator reliability has improved and the industry as a whole is maintaining reliability above the chosen target levels. NUMARC maintains that these efforts and results are adequate and that the prescriptive guidance contained in R.G. 1.9, Rev. 3, is unwarranted.

We believe that the commitments of the licensees to monitor and maintain diesel generator reliability above the chosen target levels and the industry initiatives are sufficient to ensure acceptable diesel generator reliability under the Station Blackout Rule. If plants fall below the target levels, these plants should be identified and corrective actions will be taken.

We recommend that the prescriptive guidance contained in R.G. 1.9, Rev. 3, Sections C.6-2 through C.6-7 be removed, along with the related figures and tables. In addition, the staff should not issue a 50.54(f) letter to impose adoption of R.G. 1.9, Rev. 3.

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

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Member Harold W. Lewis

First, I don't see the problem this program is supposed to solve. Everyone seems to agree that diesel reliability is good and improving, and that each diesel failure should be analyzed for root cause, to reduce the likelihood of recurrence. The remaining issue is the relevance of the threshold values.

Clearly, failure experience is an indicator of the underlying reliability -- the question is how to use the data. I am surprised that such a trivial measure as a collection of arbitrary threshold values has been chosen. Once the failure data have been collected, it is no harder to make full use of the data, by calculating a set of confidence limits on the underlying reliability. Such a procedure makes optimal use of the data, and can be recalculated after each attempt to start, with the expenditure of a few microseconds of computer time. The trends and their significance can then be monitored. I see no excuse for throwing away data, once collected. Despite the staff assertions that this would be far more difficult, it would in fact be trivial.

References:

1. U.S. NRC Regulatory Guide 1.9, Rev. 3 (June 14, 1990), Working Draft, "Selection, Design, Qualification, Testing, and Reliability of Emergency Diesel Generator Units Used As Class 1E Onsite Electric Power Systems At Nuclear Power Plants."
2. U.S. NRC Regulatory Guide 1.9, Rev. 2 (December 1979), "Selection, Design, Qualification of Diesel-Generator Used as Standby (On-Site) Electric Power Systems at Nuclear Power Plants."
3. Nuclear Management and Resources Council, NUMARC 87-00, (Revision 1), "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout At Light Water Reactors," Appendix D, "EDG Reliability Program," May 2, 1990
4. IEEE Standard 387-1984, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," June 1984.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.155 (Task SI 501-4), "Station Blackout," August 1988.
6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.108, Rev. 1, "Periodic Testing of Diesel Generators Used As On-Site Electric Power Sysems At Nuclear Power Plants," August 1977.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

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

September 11, 1990

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Carlyle Mi[illegible]son *Carlyle Michelson*
Chairman, [illegible]

SUBJECT: PROPOSED PRIORITY RANKINGS OF GENERIC ISSUES: SIXTH GROUP

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we discussed the priority rankings proposed by the staff for a group of generic issues identified in Table A, attached. Our comments are contained in the following attachments:

- Attachment 1 lists those issues for which we agree with the priority rankings proposed by the staff.
- Attachment 2 includes those issues for which we agree with the proposed priority rankings, but have comments.
- Attachment 3 identifies the generic issue for which we disagree with the proposed priority ranking.

We request that the NRC staff provide written responses to the comments included in Attachments 2 and 3.

We will continue our review of proposed priority rankings for additional generic issues when they become available.

Attachments:
As stated

TABLE A

GENERIC ISSUES REVIEWED BY THE ACRS DURING THE 365TH, SEPTEMBER 6-7, 1990 MEETING

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
15	Radiation Effects on Reactor Vessel Supports	HIGH	Memorandum from E. S. Beckjord for R. W. Houston, February 8, 1989
43	Reliability of Air Systems	HIGH	Memorandum from E. S. Beckjord for B. W. Sheron, April 11, 1988
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	Memorandum from E. S. Beckjord for B. W. Sheron, June 8, 1988
62	Reactor Systems Bolting Applications	The safety concerns of this issue are being addressed in the resolution of Generic Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"	Memorandum from E. S. Beckjord for R. W. Houston, August 18, 1988
63	Use of Equipment not Classified as Essential to Safety in BWR Transient Analysis	DROP	Memorandum from E. S. Beckjord for B. M. Morris, February 26, 1990
71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	LOW	Memorandum from E. S. Beckjord for B. M. Morris, February 26, 1990
81	Impact of Locked Doors and Barriers on Plant and Personnel Safety	DROP	Memorandum from E. S. Beckjord for B. M. Morris, February 27, 1990
95	Loss of Effective Volume for Containment Recirculation Spray	Nearly Resolved	Memorandum from E. S. Beckjord for T. E. Murley, February 27, 1990

TABLE A (Cont'd)

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
96	RHR Suction Valve Testing	The safety concerns of this issue have been integrated into the resolution of Generic Issue 105, "Interfacing Systems LOCA at LWRs"	Memorandum from E. S. Beckjord for W. Minners, February 26, 1990
104	Reduction of Boron Dilution Requirements	DROP	Memorandum from E. S. Beckjord for R. W. Houston, August 18, 1988
107	Main Transformer Failures	LOW	Memorandum from E. S. Beckjord for B. M. Morris, February 26, 1990
109	Reactor Vessel Closure Failure	DROP	Memorandum from E. S. Beckjord for B. M. Morris, February 27, 1990
117	Allowable Outage Times for Diverse Simultaneous Equipment Outages	DROP (The safety concerns of this issue have been addressed in the Technical Specification Improvement Program)	Memorandum from E. S. Beckjord for T. E. Murley, February 26, 1990
125.I.5	Safety Systems Tested in all Conditions Required by the Design Basis	DROP	Memorandum from E. S. Beckjord for R. W. Houston, November 3, 1988
125.II.11	Recovery of Main Feed-Water as Alternative to Auxiliary Feedwater	DROP (The safety concerns of this issue are being addressed in the resolution of Generic Issue 124, "Auxiliary Feedwater System Reliability")	Memorandum from E. S. Beckjord for RES Division Directors, June 22, 1988

TABLE A (Cont'd)

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
129	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	DROP	Memorandum from E. S. Beckjord for B. M. Morris, February 26, 1990
131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants	MEDIUM (The safety concerns of this issue will be addressed through the IPEEE Program)	Memorandum from E. S. Beckjord for R. W. Houston, July 31, 1989
137	Refueling Cavity Seal Failure	DROP	Memorandum from E. S. Beckjord for B. M. Morris, May 7, 1990
139	Thinning of Carbon Steel Piping in LWRs	Regulatory Impact Issue	Memorandum from E. S. Beckjord for T. E. Murley, November 4, 1988
140	Fission Product Removal Systems	DROP (The safety concerns of this issue have been addressed in the Severe Accident Program with changes to the Standard Review Plan)	Memorandum from E. S. Beckjord for B. M. Morris, February 26, 1990
141	Large Break LOCA with Consequential SGTR	DROP	Memorandum from E. S. Beckjord for B. M. Morris, May 2, 1990
142	Leakage Through Electrical Isolators in Instrumentation Circuits	MEDIUM	Memorandum from E. S. Beckjord for W. Minners, June 20, 1990
B-31	Dam Failure Model	DROP	Memorandum from E. S. Beckjord for R. W. Houston, February 8, 1989

TABLE A (Cont'd)

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
D-2	ECCS Capability for Future Plants	DROP (The safety concerns of this issue will be addressed in the Severe Accident Policy Implementation Program)	Memorandum from E. S. Beckjord for R. W. Houston, October 7, 1988
III.D.1.1 (2)	Review Information on Provisions for Leak Detection	DROP (See below)	Memorandum from E. S. Beckjord for RES Division Directors, September 1, 1988
III.D.1.1 (3)	Develop Proposed System Acceptance Criteria	DROP (The safety concerns of III.D.1.1(2) and III.D.1.1(3) have been addressed in NUREG-0737 and in the proposed Broad Scope Amendment to GDC-4)	

ATTACHMENT 1

LIST OF GENERIC ISSUES FOR WHICH THE ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF

GENERIC ISSUE NO.	TITLE
15	Radiation Effects on Reactor Vessel Supports
43	Reliability of Air Systems
57	Effects of Fire Protection System Actuation on Safety-Related Equipment
62	Reactor Systems Bolting Applications
63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis
71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety
95	Loss of Effective Volume for Containment Recirculation Spray
104	Reduction of Boron Dilution Requirements
107	Main Transformer Failures
109	Reactor Vessel Closure Failure
117	Allowable Outage Times for Diverse Simultaneous Equipment Outages
125.I.5	Safety Systems Tested in All Conditions Required by the Design Basis
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater
131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants
137	Refueling Cavity Seal Failure
139	Thinning of Carbon Steel Piping in LWRs
140	Fission Product Removal Systems

141	Large Break LOCA with Consequential SGTR
142	Leakage Through Electrical Isolators in Instrumentation Circuits
B-31	Dam Failure Model
III.D.1.1(2)	Review Information on Provisions for Leak Detection
III.D.1.1(3)	Develop Proposed System Acceptance Criteria

ATTACHMENT 2

LIST OF GENERIC ISSUES FOR WHICH THE ACRS AGREES WITH THE PROPOSED PRIORITY RANKINGS BUT WITH COMMENTS

Generic Issue No: 96

Title: RHR Suction Valve Testing

Proposed Priority Ranking: The safety concerns of this issue have been integrated into the resolution of Generic Issue No. 105, "Interfacing Systems LOCA at LWRs."

ACRS Comment: We agree with the staff's proposal to integrate the safety concerns of this issue into the resolution of Generic Issue 105. We believe that failure of both RHR suction valves may not be very likely, but the consequences of such an occurrence could be severe. Results of the Indian Point and Zion PRAs revealed that the dominant interfacing systems LOCA (ISLOCA) events involved the failure of RHR suction valves. Therefore, special attention should be given to this dominant contributor to ISLOCA in the resolution of Generic Issue 105.

Generic Issue No: 129

Title: Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling

Proposed Priority Ranking: DROP

ACRS Comment: We agree with the proposed priority ranking for this generic issue. However, we believe that this issue should receive attention in the PRA studies now under way to investigate the risks from events that occur during shutdown operations.

Generic Issue No: D-2

Title: ECCS Capability for Future Plants

Proposed Priority Ranking: DROP (The safety concerns of this issue will be addressed in the Severe Accident Policy Implementation Program.)

ACRS Comment: We agree with the staff's proposal to address the safety concerns of this issue in the Severe Accident Policy Implementation Program. However, we offer the following comments.

The ECCS design for future plants is now based on Appendix K to 10 CFR Part 50 in its unrevised form. In light of what is known today, and given the industry's calculational capabilities, there is no reason to continue to use the unrevised form of Appendix K. Implementing the Commission's Severe Accident Policy will not change the fact that the ECCS will be designed and operated according to a set of non-physical rules rather than the best tools available. Overall safety enhancement by implementation of the Commission's Severe Accident Policy may well be compromised as a result.

ATTACHMENT 3

GENERIC ISSUE FOR WHICH THE ACRS DISAGREES WITH THE PROPOSED PRIORITY RANKING

Generic Issue No: 81

Title: Impact of Locked Doors and Barriers on Plant and Personnel Safety

Proposed Priority Ranking: DROP

ACRS Recommendation: Be Reanalyzed

Reasons: The risk "calculation" to support the "Drop" priority ranking for this generic issue is worthless. The staff argument to drop this issue is as follows:

1) There is a 99% probability of success in penetrating a locked barrier within an hour, and the probability dependence on time is an exponential, $1-\exp(-Kt)$.

2) The probability of core melt, given a failure to penetrate the barrier in an hour is unity, and its dependence on time is a power law, At^n, where $n>o$.

3) The overall probability of core melt is the product of these two, and is maximized by assuring that they are equal to each other, and that their slopes are equal and opposite.

4) The maximum probability is then 3.4×10^{-2}, at 22 minutes.

There is no justification for either the number or the functional dependence in (1) or (2). The procedure in (3) is mathematically incorrect.

Therefore, no credibility can be assigned to the conclusion in (4), on which the rest of the argument rests.

We have seen no evidence that the recommendation to drop is correct, and it is unsupported by the purported analysis. It may be true, but that has not been demonstrated. We recommend that the analysis be done correctly and resubmitted.

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

August 16, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: NRC RESEARCH ON ORGANIZATIONAL FACTORS

During the 364th meeting of the Advisory Committee on Reactor Safeguards, August 9-11, 1990, we reviewed, at the request of the NRC staff, the Commission's program of research related to organizational factors. Our Subcommittee on Human Factors also reviewed this matter during a meeting on July 31, 1990. During these meetings, we had the benefit of discussions with representatives of the NRC staff and its contractor, Brookhaven National Laboratory. We also had the benefit of the document referenced.

The NRC research program on organizational factors is intended to provide a scientific basis for improving the organizations responsible for operating nuclear power plants. The Commission has expressed concerns about the feasibility of such research and has asked to be briefed on the status of the program. We recognize the reasons for these concerns; the issues are difficult and are outside the mainstream experience of the NRC and the industry. This does not mean the issues should be ignored since they are of vital importance to nuclear power plant safety.

The Commission, the ACRS, and the nuclear power industry have recognized for the past several years that the quality of management associated with nuclear power plant operations is of cardinal importance to plant safety. During our August meeting, Dr. Herbert Kouts, the Chairman of the Special Committee to Review the Severe Accident Risk Report (NUREG-1150), summarized the results of that Committee's review. In response to a question, he noted his support for continuing NRC research on human reliability analysis, in particular research on the influence of organizational factors.

An important component of good management is an effective plant organization. Little quantitative basis exists for optimizing plant organizational design with respect to safety. This contrasts with the comprehensive technical bases that support many other aspects of nuclear power plant safety and design.

Under the present research program, the staff and its contractors are studying organizations ranging in scope and size from the total licensee force at a nuclear power plant site to the shift crews and smaller teams that perform essential functions of operation and maintenance. Depending on the results of this work, the program may be expanded at a later time to study the effect of utility and other organizations external to the plant. It has been recognized that complex nuclear power facilities are operated and maintained by teams of people, not by individuals. Therefore, something more than training and licensing of individual operators is necessary to ensure plant safety.

The research program described to us by the staff appears to be focused on agency needs and can make a contribution to future improvements in the effectiveness of nuclear power plant organizations. We do have a concern that the research program seems to be directed toward the need to consider operator performance in PRAs in a more quantitative manner. This is a desirable ultimate goal; however, we believe that more emphasis should be placed on communicating to nuclear power plant licensees the insights developed on effective managerial approaches.

Continued support and encouragement for this research program from the Commissioners and the NRC staff management will be necessary. The research staff and its contractors are undertaking a difficult and pioneering effort. We will follow progress of the program with interest.

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

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I was less impressed than my colleagues. All of American industry is concerned about the effect of organization on productivity and effectiveness - courses are taught, books are written, etc. I don't believe the need is for research, but for application of what is known to the NRC's regulatory problems. Not only the industry, but NRC itself, could benefit. No one doubts the importance of the subject.

Reference:
Draft SECY paper to the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Organizational Factors Research Progress Report (Predecisional), transmitted by Memorandum dated July 5, 1990 from Tom Ryan, Office of Nuclear Regulatory Research, NRC, to Herman Alderman, ACRS

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

February 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: ACRS REVIEW AND EVALUATION OF NUCLEAR POWER PLANT OPERATING EXPERIENCE

In your letter dated January 12, 1990, you requested a report discussing ACRS reevaluation of the Committee's role in the review and evaluation of nuclear power plant operating experience.

During our January 11-12, 1990 meeting, we reviewed the existing ACRS subcommittee structure with particular emphasis on identifying issues requiring increased ACRS involvement. Several new subcommittees were established and the scopes of some existing subcommittees were modified to reflect the Committee's perception of its future activities.

Attached is a list of the current and recently disbanded ACRS subcommittees. You will note that many of the subcommittees are involved with some facet of plant operating experience as part of their charters. Five subcommittees have specific assignments for review of operating experience:

o A Plant Operations Subcommittee was established to act as the lead subcommittee in the area of plant operating experience. This subcommittee will review selected nuclear power plant operating events and the short-term actions associated with these events, including interfacing with the NRR Division of Operational Events Assessment. The subcommittee will consider, as appropriate, activities such as the LER, SALP, Diagnostic Evaluation Team, AIT, and IIT processes; operational quality; operator performance; and Technical Specification issues.

- A Systematic Assessment of Experience Subcommittee has the responsibility for the long-term and generic implications of operating experience, including interfacing with AEOD and INPO on activities regarding long-range assessment of operating experience, reviewing lessons learned from AIT and IIT reports, comparing PRA input/results to actual operating experience, and reviewing the development of operational performance indicators.

- An Adopted Plants Activities Subcommittee was established to handle procedural and resource issues and member assignments associated with the ACRS adopted plants program. This program, which was established in 1987, involves the assignment of a number of operating plants to each ACRS member. The member receives licensing and inspection correspondence and SALP reports on his plants, visits the plants as appropriate, and reports items of interest to the ACRS.

- An International Activities Subcommittee was established to coordinate all activities on foreign reactors and interface with AEOD and GPA on international activities. Subcommittee responsibilities include the review and evaluation of foreign nuclear power plant operating experience.

- The Regional Activities Subcommittee meets periodically with each Regional Administrator and his staff to review the various inspection and enforcement programs in the Regions. A complete round of these meetings has taken place and a second round will commence this year.

In addition to these subcommittee activities, the ACRS staff receives and reviews Weekly Information Reports; NRC bulletins, notices, and generic letters; AEOD reports; performance indicator reports; preliminary notifications; the monthly compilation of LERs; and Headquarters and Regional Office daily reports. Committee members receive certain of this information depending on their interests. This information provides input to the Committee's process for establishing subcommittee work assignments and requesting staff briefings on operating experience issues.

In summary, we believe that ACRS has the appropriate mechanisms in place to review and evaluate U.S. and foreign nuclear plant operating experience and to advise the Commission in this important area.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Attachment:
List of Current and Recently
Disbanded ACRS Subcommittees

ATTACHMENT

LIST OF CURRENT AND RECENTLY DISBANDED
ACRS SUBCOMMITTEES

NEW SUBCOMMITTEES

Adopted Plants Activities
Defueling/Fuel Pool Storage
International Activities
Plant License Renewal
Plant Operations
TVA Plant Licensing/Restart

PREVIOUSLY ESTABLISHED SUBCOMMITTEES

Comanche Peak Units 1 & 2
Computers in Nuclear Power Plant Operations
Seabrook Nuclear Plant Unit 1
AC/DC Power Systems Reliability
Advanced BWRs (GE)
Advanced PWRs (W & CE)
Advanced Reactor Designs
Auxiliary & Secondary Systems
Babcock & Wilcox Reactor Plants
Containment Systems
Core Performance
Decay Heat Removal Systems
Extreme External Phenomena
FTOL Conversions
General Electric Reactor Plants
Generic Items
Human Factors
Improved LWRs
Instrumentation and Co[illegible] Systems
Integrated Safety Asses[illegible] Program
Maintenance Practices a[illegible]ocedures
Materials & Metallurgy
Mechanical Components
Naval Reactors
Non-Power Reactors
Occupational and Environmental Protection Systems
Planning and Procedures
Probabilistic Risk Assessment
Quality and Quality Assurance in Design and Construction
Regional Programs
Regulatory Activities
Regulatory Policies and Practices
Reliability Assurance
Safeguards and Security
Safety Philosophy, Technology and Criteria

Safety Research Program
Severe Accidents
Structural Engineering
Systematic Assessment of Experience
Thermal Hydraulic Phenomena
Westinghouse and Combustion Engineering Reactor Plants

SUBCOMMITTEES WHICH HAVE RECENTLY BEEN DISBANDED

(Tasks have been absorbed into either new or other previously established subcommittees)

Bellefonte Plant Units 1 & 2
Limerick Unit 2
Watts Bar Units 1 & 2
ACRS Bylaws
Consideration of International Operating Experiences
Member Nominations
On-Site Fuel Storage
Plant Operating Procedures

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

February 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: COHERENCE IN THE REGULATORY PROCESS

In our reports to you of November 24, 1989 (which also lists our earlier reports) and December 21, 1989, we have discussed a variety of aspects of the coherence problem -- the problem of assuring that all elements of the NRC pull in the same direction in the regulation of nuclear power, a direction provided by the Commission itself. These reports have generally dealt with symptoms of incoherence -- the most recent was about the internal use of SALP ratings. Here we would like to take a more global view of the coherence problem, leading in the end to a recommendation for a next step.

It is almost as if the NRC were created to be incoherent. There are five Commissioners and five statutory Offices. There are many Branches and five Regional Offices, with a kind of matrix management tieing it all together. Regulatory power is spread throughout, resulting in a melange of technical positions, regulatory guides, generic letters, policy statements, undocumented pressures, enforcement actions, etc. The mechanisms for providing incentive to the various elements of the staff to test their actions in the light of Commission objectives are inadequate. Indeed those objectives are not always easy to determine, for reasons that need no elaboration here. This is not to say that anyone is deliberately misbehaving, only that too many are free to proceed in the light of their own best judgment.

We have long argued that the best way to test the effectiveness of the regulatory process is to measure the results in terms of the Commission's Safety Goals, and we do not depart from that position here, but a performance measure is not a coherence measure. The latter has to do with efficiency, clarity, and ultimately, acceptability of the process.

In our November 24 report on this subject we emphasized that the coherence problem can be divided into many categories -- it is not a neat subject. The Commission itself can and should make its policy statements and other issuances as unambiguous as possible (we know that is not easy; we often fail ourselves), so as to minimize opportunities for misinterpretation. Also, as mentioned in that report, many of the examples lie within the province of the EDO, and he should be aware of his responsibility to keep the various offices working toward the same ends. Perhaps his own staff needs expansion. But the real tests of coherence lie in the NRC's interactions with the outside world, and we doubt that only internal modifications can solve these problems, although we believe improvements could be made. We are not prepared to recommend reorganization of the NRC, though that is one of the options available to you. Certainly, incentives for lateral communication would be helpful.

We do not believe coherence can be proclaimed from above. Not only is the effect of proclamations attenuated as they penetrate any organization, but high-level policies are necessarily imprecise. Not all ramifications or interpretations of a policy statement can be foreseen, and coherent policies have to be molded in use. It is the body of regulatory practice that is in question here, much of it in the form of corporate memory and lore, and the job at each level is to provide sufficient guidance and incentive to make it possible (and desirable) for the next level to function consistently with the global policies. Above all, the governing policy guidance must be simple, clear, and understandable to both regulator and regulatee.

How is coherence approached elsewhere? One necessary ingredient appears to be feedback, through which interpretations of policy are constantly tested against the policies themselves, not in every case but through a sampling process that, in the end, leads to a more coherent structure. The guiding law of the land is the Constitution, embodying our principles of government. The real law of the land, however, is the enormous body of case law generated by innumerable court decisions, each reviewable, and some in fact reviewed, by the next level of appellate court. Thus the regulatees, in this case the population, have a set of recourses that can bring any rule or ruling to a test of its coherence with the guiding principles. Further, and most important, those who do the testing are not those who make the rules, so there is at least the perception that there is a genuinely unbiased feedback process. The founders were careful to include this in the system. In addition, feedback loops need not be end-to-end; intermediate loops are also helpful.

There are many examples of this process in other areas. A taxpayer who feels mistreated by the Internal Revenue Service can appeal within the system, but can in the end go to the Tax Court, an entirely independent forum. A pilot denied his or her license by the Federal Aviation Administration has the right to appeal to the National Transportation Safety Board, an independent agency, whose ruling is final. In each of these there is some risk, but the constant feedback provided by external review helps to create a body of case law that is under continuous testing for coherence. This is not true in the nuclear business, where the only external review is in the courts, and their primary mission is not coherence in the regulatory process. The only appeal from a Regional decision (for example) is within the system, and we all learn early that it is unwise to complain about someone who has power over you, unless you're sure you'll win.

All engineers recognize that complex systems are better controlled by feedback than by blind input -- one measures the errors and corrects the input accordingly. The key is the ability to make objective measurements through a separate sensing system.

What appears to be needed in our case is a mechanism through which frequent testing of the body of "case law" against the guiding principles laid down by the Commission is made possible. To be credible and effective, that job cannot be assigned entirely to the Commission staff. The current situation is analogous to one in which there is a constitution (Commission policies), a body of law (letters, guides, enforcement actions, rules), but no courts.

In general, those with the most to gain by coherent regulation are the regulatees (and of course the rest of us, because safety will benefit), and they would be in a better position to seek coherence if they could do so without fear of retaliation. It is the fear of being taken to court that serves to constrain police forces -- the constraints in our case are entirely internal.

This kind of feedback solution has been used in many places. Governments and police forces have courts; factories have grievance committees; some agencies have ombudsmen for employee complaints, though these usually have no power. The NRC has nothing comparable.

We believe the ultimate solution to the coherence question must include the provision of an adequate feedback mechanism. To be sure, you have made any number of commendable requests to the regulated community to come forward with complaints, but less has come of it than might have been hoped. Even if more had happened, this would still have been symptomatic treatment of the problem, and we believe that a mechanism in place is required. Some of us believe that, in the end, only an external Nuclear Safety Board can help, while others believe that great strides can be made within the NRC itself. However, just as we are not prepared to recommend reorganization of the NRC, we do not suggest what form the feedback mechanism should take.

We do recommend that possible means for achieving the objects stated above be explored, and doubt that it would be wise to simply ask the staff (or us) to do the job. We think it would be entirely appropriate, given the importance of the issues, to take a major initiative by asking some respected outside group to explore the subject, and to lay out the feedback options available to the country, even if they require legislation. Such a study group could be chartered by the NRC -- there are precedents -- and should include representation from the affected industry. The National Academy of Sciences has done such things, or it could be an entirely free-standing operation. The result should not be a specific recommendation, but a list of options and analyses, which could then be freely debated within the interested community. This is a complex subject, and we do not think it should be resolved by hip shot. We also do not think it should be neglected, since the effectiveness of the regulatory process is at issue.

Additional comments by ACRS Members Carlyle Michelson, Chester P. Siess, and Charles J. Wylie are presented below.

Sincerely,

Harold W. Lewis
Acting Chairman

<u>Additional Comments by ACRS Members Carlyle Michelson, Chester P. Siess, and Charles J. Wylie</u>

If there is a problem with coherence in the regulatory process, we do not believe that it has been identified and characterized in this report with sufficient clarity to support a recommendation that the NRC charge some outside group to explore it. We agree that there have been examples of inadequate integration of regulatory staff activities, sometimes serious, but it should not require an outside panel to tell management how to correct such deficiencies. If the ACRS believes that there is a coherence problem beyond the capability of the Commission to highlight and correct, then it should clearly articulate the problem before suggesting that the ultimate solution must include provisions for an adequate feedback mechanism and asking some outside group to lay out the feedback options. There are other portions of this letter to which we would take exception; but unless the ACRS can define the problem that needs to be fixed, they may not be worthy of mention. It is our observation that the agency knows its responsibilities and has been successful in carrying out its mission.

October 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

We have your memorandum of August 1, 1990, dealing with legal services for the Advisory Committee on Reactor Safeguards. We wish to comment on the implication in your memorandum that the ACRS role is to provide "scientific and technical" advice to the Commission and to spell out a bit more carefully the basis for the position taken by the ACRS in a letter to you dated July 17, 1990.

The basic documents that specify the ACRS duties are Section 29 of the Atomic Energy Act, as amended, and the provisions contained in 10 CFR 1.13. You imply in your memorandum that these documents define the ACRS role as one of giving "scientific and technical advice" to the Commission, but the fact is that no such language is contained in either. To the contrary, both documents refer to advice on a variety of safety-related matters. Until now no Commission and no Chairman have defined limits to this assignment.

Although the July 17 letter called this matter to your attention, it now appears that it would be helpful to explain more carefully just why it is important to reactor safety that we have the freedom to explore (including the use of appropriate consultants) all those aspects of a safety-related question that we deem important. The point made in the July 17 letter is that independence only on narrowly technical matters is unduly limiting.

The nub of the issue is that reactor safety is a complex mix of technical, procedural, human, and legal matters. For any given safety question one or another of these factors may dominate, and to limit the areas of investigation in advance is to seriously impair the ability of the Committee to function in its statutory role. Perhaps some examples will help.

- In 1986 the interpretation of the backfit rule was a pressing issue, involving both the extent to which a cost-benefit analysis could be required as justification for a backfit, and the definition of adequate protection. The Commission had already received a report from OGC on these matters, but the Committee felt that, in its role as an independent advisor to the Commission, it required a separate analysis. The Committee then engaged an outside law firm to study the issues on its behalf, and that study materially contributed to its understanding. In this case, the legal issues were inseparable from the technical ones.

- Though we have yet to report to you on this, we have recently had a series of briefings on the criteria the staff has used to order a plant shut down and to permit it to restart. This discussion has raised, in some of our minds, serious questions about accountability for each of these decisions. Both for shutdown and for restart, the staff criteria were highly personal and subjective in areas (like "management culture") that lack explicit standards. Whether it is in the interests of nuclear safety for the licensee to be forced to simply placate the staff under these conditions is at best questionable. Certainly the staff has limited expertise in such areas.

These are two (of many that could have been furnished) examples of important safety-related matters, which are not narrowly "scientific and technical." The Committee is required by both law and conscience to advise you about all aspects of safety-related matters, without topical constraints. This will occasionally require that we seek outside consultation on a variety of subjects when a second opinion seems appropriate, even though the advice available from your staff may well be competent. (Such outside consultation may well involve legal matters.) After all, it is the staff that advises you, and our independence is illusory if we are confined to that same staff for our own inputs.

Once more we ask you to take these matters seriously -- they go to the heart of the relationship between the Commission and the Committee. We do not raise them lightly, and urge you to reconsider the position taken in your memorandum of August 1, 1990.

Additional comments by ACRS Members Carlyle Michelson and Charles J. Wylie, and by Chester P. Siess are presented below.

Sincerely,

Harold W. Lewis
Acting Chairman

<u>Additional Comments by ACRS Members Carlyle Michelson and Charles J. Wylie</u>

It is our position that Chairman Carr's memorandum of August 1, 1990, constitutes an adequate reply to the ACRS letter of July 17, 1990. We believe that the ACRS is not constrained in pursuit of its responsibilities as defined by the Atomic Energy Act and by Federal regulations. If it should require legal assistance concerning a specific matter, the Office of the General Counsel is ready and willing to support such a need. If the Committee should feel that independent legal assistance is essential, the Commission has ensured that such a need can be brought to its attention for resolution. To our knowledge, the Committee has never been encumbered in its efforts to find and retain outside scientific or technical assistance. It is our view that this matter has already achieved a proper closure and should be dropped.

Additional Comments by ACRS Member Chester P. Siess

I cannot agree with my colleagues that my ability to provide advice to the Commission on matters of reactor safety is seriously impaired by anything you wrote in your memorandum of August 1, 1990.

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

June 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: MODIFIED ENFORCEMENT POLICY FOR HOT PARTICLE EXPOSURES INCORPORATING THE RECOMMENDATIONS OF NCRP REPORT NO. 106

During the 362nd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 1990, we discussed with the NRC staff its currently proposed enforcement policy (contained in SECY-90-169, Modified Enforcement Policy for Hot Particle Exposures - Revision to Incorporate Recommendations made in NCRP Report No. 106). This proposed policy addresses the use of enforcement discretion in cases involving occupational dose to the skin of radiation workers from exposure to radiation from hot particles. We also had the benefit of the referenced document.

The Committee previously commented on this matter in its report dated May 9, 1989 that presented the results of our review of the staff's originally proposed generic letter and draft interim standard on occupational dose resulting from hot particle exposure. In this report the Committee recommended against the issuance of that generic letter and interim standard on the basis that the letter and standard did not reflect the recommendations of a then-pending National Council on Radiation Protection and Measurements (NCRP) report on this subject. This report had been requested by the staff because of concerns that radiation workers in nuclear power plants were receiving unnecessary whole-body exposure as a result of the measures being used to protect workers against exceeding the existing limits for skin exposure. These limits, which are based on the exposure of large areas of the skin, were recognized to be extremely conservative when applied to hot particle exposure.

The Committee recommended that staff senior management "... take an active role in effecting a timely resolution of remaining outstanding issues with NCRP so that its report may be published." The Committee further recommended that the staff "... then develop on an expedited basis an interim standard based on the NCRP recommendations." or "To the extent the standard differs from the NCRP recommendations, the staff's reasons for such modifications should be clearly and completely documented." In addition, the

Committee recommended that "... the staff concurrently move ahead with its planned revision of 10 CFR Part 20 on this subject."

On December 11, 1989, the staff proposed a second modified enforcement policy for hot particle exposures (SECY-89-370, Modified Enforcement Policy for Hot Particle Exposures) because the NCRP report had not been published in final form. In the Commission's March 8, 1990 Staff Requirements Memorandum (SECY-89-370), the staff was asked to revise this proposed policy to use the recommendation of NCRP as a basis for the policy or provide an alternative approach.

On December 31, 1989, NCRP published its Report No. 106, Limit for Exposure to "Hot Particles" on the Skin, in final form. The report had been subjected to several rounds of comments by the staff, NUMARC, and others before its publication.

We have reviewed the staff's latest modified interim enforcement policy for hot particle exposure based on the recommendations of the NCRP report (Enclosure 2 to SECY-90-169). We find this interim enforcement policy to be an acceptable approach until a new limit can be established by revision of 10 CFR Part 20. We believe that this rulemaking on the hot particle issue should be given high priority by the staff. We continue to be impressed by NCRP's expertise in dealing with complex technical issues such as this and with their expertise in radiation protection matters in general.

During our meeting on June 7-9, 1990, we also discussed a number of the "technical issues" raised in Eric S. Beckjord's letter to Thomas E. Murley (Enclosure 3 to SECY-90-169) with the meeting participants. Based on the comments that were presented during our June meeting, we believe that Mr. Beckjord's letter presented an incomplete view of the current hot particle situation. Some examples are as follows:

1. Mr. Beckjord's letter does not mention the EPRI-sponsored "pig studies" recently completed by Pacific Northwest Laboratory. These studies indicate that the position taken by NCRP is a conservative one with respect to fission product hot particles and an extremely conservative one with respect to cobalt-60 hot particles. These are the two prevalent types of hot particles found in operating nuclear power plants.

2. The NCRP endpoint of "deep ulceration" as a basis for its limit should have been placed in perspective in Mr. Beckjord's letter. We understand that this limit implies, at most, a tiny ulcer that would quickly heal. Such an ulcer, while clinically detectable, would probably not be noticed by the radiation worker.

3. Mr. Beckjord's letter expresses the staff's interpretation of the views of two of five members of the International Commission on Radiation Protection and Measurements (ICRP) Task Group on Skin (Drs. Monty Charles and John Hopewell, two British researchers). The other three members are from the U.S. and were members of the NCRP Scientific Committee that prepared NCRP Report No. 106.

4. Dr. John Hopewell (see 3 above) participated in the EPRI-sponsored meeting on hot particles held in Richland, Washington, last June. Representatives from ACRS and RES also attended this meeting. During the meeting, Dr. Hopewell was asked about the tradeoff he would make between additional whole-body exposure to prevent a given hot particle exposure. His view was that the mission of the ICRP Task Group was to recommend a limit that would conservatively prevent non-stochastic effects without regard to the whole-body exposure that might be incurred to prevent hot particle exposure. He stated that this kind of tradeoff (which NCRP had made in its recommendations) would be considered by the full ICRP.

5. We were very surprised by the number of "technical issues" contained in Mr. Beckjord's letter, given the extensive dialogue that has occurred between the staff and the NCRP over the past several years on this matter.

The above comments on Mr. Beckjord's letter tend to reinforce the Committee's view stated in its report dated May 9, 1989 that "... staff senior management take an active role in effecting a timely resolution of remaining outstanding issues with NCRP ...." We believe that it is important that a good working relationship be established and maintained between the staff and the NCRP.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
SECY-90-169, Memorandum dated May 11 1990 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Modified Enforcement Policy for Hot Particle Exposures - Revision to Incorporate Recommendations Made in NCRP Report No. 106, with Enclosures (Predecisional)

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

April 26, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 358th, 359th, and 360th meetings of the Advisory Committee on Reactor Safeguards, February 8-10, March 8-10, and April 5-7 and 18-19, 1990, we discussed with representatives of the NRC staff the staff's positions and recommendations concerning the evolutionary light water reactor (ELWR) certification issues contained in SECY-90-016 (Ref. 1). During some of these meetings, we had the benefit of discussions with representatives of the Electric Power Research Institute (EPRI) and the General Electric Company. We also had the benefit of the documents referenced.

We were told by the staff that the positions for which they are seeking Commission approval are described in the underlined portions of the enclosure to SECY-90-016, entitled "Evolutionary ALWR Certification Issues." Unless indicated otherwise, our comments relate to these staff positions. Our comments and recommendations on the staff positions are given below.

I. GENERAL ISSUES

1. Evolutionary LWR Public Safety Goals

The NRC staff has concluded that the quantitative goals submitted for Commission consideration in draft SECY-89-102 (Ref. 2) are acceptable for ELWRs. The staff notes that both public safety goals in the EPRI ALWR Requirements Document (Ref. 3) and the ABWR Licensing Review Basis Document (Ref. 4) are considerably more restrictive than the large-release guideline defined in draft SECY-89-102. The staff further notes that additional Commission guidance on quantitative safety goals will assist the staff in its continuing assessment of ELWRs.

We believe, as stated in our previous reports (e.g., ACRS report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs, dated July 20, 1988), that the Commission's Safety Goal Policy is appropriate guidance for regulatory decisions relating to ELWRs, other advanced reactors, and the operating plants. We regard it as not inappropriate that applicants should work to tighter standards when it serves their purposes, but we do not believe it is appropriate that the NRC should require such standards. In its Safety Goal Policy the Commission, in effect, said it would regulate to a level of safety that is adequate, not to the highest level that is possible.

2. Source Term

This issue is dealt with by a proposal to assure that evolutionary designs meet the requirements of 10 CFR 100 (Reactor Site Criteria). The requirements of this regulation include a limit on doses experienced by an individual at the exclusion area boundary, and at the boundary of the low population zone during the course of an accident. In calculating these doses, the instructions in 10 CFR 100 prescribe that the fission products released to the containment must be those which would be expected from accidents which "result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." For plants currently operating, regulatory guides have delineated specific, but somewhat arbitrary, quantities of fission products that are acceptable to the staff in calculating the leakage from containment and the resultant doses at the specified boundaries.

In contrast, for the ELWRs, the staff proposes to explore the specification of a source term on a case-by-case basis, rather than using the arbitrary source term prescribed in the past. Since the issue of siting of these plants is not yet resolved, and since revisions to 10 CFR 100 are being considered, there may be no alternative to proceeding as the staff proposes, however awkward it may seem.

However, we can make no informed judgment concerning the appropriateness of the procedure until we know more about the criteria to be used in the selection of a source term, and the results of its application.

II. PREVENTATIVE FEATURE ISSUES

3. Anticipated Transients Without Scram (ATWS)

The staff recommends that the Commission approve the staff's position that diverse scram systems be required for the ELWRs.

It appears to us that a design that can ride out an ATWS without serious damage is feasible for PWRs and is preferable to a scram system with diverse logic, which has a reliability calculable, at best, with large uncertainty. We recommend that the staff permit demonstration that the consequences of an ATWS are acceptable as an alternative to a diverse scram logic. The uncertainty in such a demonstration is probably considerably less than that in demonstrating that the contribution of an ATWS to risk is made acceptable by installation of a diverse scram logic system.

4. Mid-Loop Operation

We have been told previously of evidence that events initiated during mid-loop operations may be major contributors to risk in PWRs. However, shutdown operations are generally not accounted for in PRA studies, such as those reported in NUREG-1150 (Ref. 5), so the risks are not well quantified. For the operating plants, this issue has been dealt with through resolution of Generic Issue 99 (Improved Reliability of RHR Capability in PWRs). For the ELWRs, the staff recommends that PWR applicants propose design features to ensure high reliability of the shutdown decay heat removal system.

We agree with the staff's proposal, but recommend that more specific requirements be considered for mid-loop operation:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low-liquid levels at the junction of the DHR system suction lines and the RCS
- Provisions to ensure availability of reliable systems for decay heat removal
- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the DHR system suction lines and the RCS
- Provisions for maintaining containment closure or for rapid closure of containment openings

5. Station Blackout

The Station Blackout Rule (10 CFR 50.63) requires that each light-water nuclear power plant licensed to operate must be able to withstand for a specified duration, and then recover from, a station blackout as defined in 10 CFR 50.2. This rule permits the utilities to submit alternative methods for coping with station blackout. This rule also states that a method based on an alternate ac power source, as defined in 10 CFR 50.2, will constitute an acceptable capability.

For the ELWRs, the staff recommends that the Commission require the installation of an alternate ac power source as the only basis taken to demonstrate compliance with 10 CFR 50.63. The staff recommends that the alternate ac source have capacity to supply power for one safety train, including one complete set of normal safe shutdown loads, and that it be of diverse design. The alternate ac power supply must be designed to serve any safety train when needed, thereby serving as an additional backup power supply for the Class IE power supplies. The staff has stated that the diversity requirement will not preclude use of diesel generators, even though diesel generators are used for the Class IE emergency power supplies.

Although taken by itself this may seem to be desirable, it has not been demonstrated that it is required to conform to the safety goal. Nevertheless, we endorse the staff's recommendation.

6. Fire Protection

The staff concluded that the fire protection issues raised through operating experience and the Individual Plant Examination for External Events (IPEEE) Program (Ref. 6) must be resolved for the ELWRs. To accomplish this, the staff is proposing that the current NRC guidance for fire protection be enhanced as described by the staff during the March 27, 1990, meeting of our Subcommittees on Extreme External Phenomena and Severe Accidents. The enhancements proposed by the staff when combined with the requirements of 10 CFR 50.48 (Fire Protection) without exception and the guidance provided by the Standard Review Plan Section 9.5.1 (Fire Protection Program) should constitute an acceptable basis for prescribing fire protection features for the ELWRs.

The proposed enhancements represent a significant improvement in physical separation requirements and in the need to consider the effects of smoke, heat, and fire suppressant migration into other areas. In particular, redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommend that the proposed enhancements include separation of environmental control systems.

The fire-risk issues that were examined in the Fire Risk Scoping Study (Ref. 7), however, are not fully addressed in SECY-90-016. They should be.

We agree with the staff's recommendation for resolution of this issue with the above caveats.

7. Intersystem LOCA

The staff's position is that designing low-pressure systems to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

III. <u>MITIGATIVE FEATURE ISSUES</u>

8. <u>Hydrogen Generation and Control</u>

The staff recommends that the ELWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of water with 100% of the fuel cladding surrounding the active fuel. (Note: This is not 100% of the fuel rod cladding, nor does it include other metal in the core which could produce hydrogen if it were heated to a red heat in the presence of steam.) There is substantial uncertainty in establishing the amount of hydrogen that might be formed in a severe accident. We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform concentrations of hydrogen greater than 10%. The EPRI ALWR Requirements Document specifies 13%. We are not aware of any experimental or analytical work that demonstrates that the detonation of hydrogen at the 10%, 13%, or some other level could damage the integrity of the containment and essential components. It is our impression that the effect, if any, is something that experts dealing with gas explosions can calculate with reasonable confidence. We suggest that the staff seek further technical information on possible effects, including stratification, before establishing a limit for the average hydrogen concentration.

9. <u>Core-Concrete Interaction - Ability To Cool Core Debris</u>

The staff proposes that the ELWR designs provide sufficient reactor cavity floor space to enhance debris spreading, and provide for quenching of the debris in the reactor cavity. Quantification of what constitutes sufficient reactor cavity floor space is still an open question, as is the means by which one quenches the core debris. The resolution of this issue will require engineering judgment as many of the physical processes are not fully understood. We agree with the staff's recommendation.

10. <u>High-Pressure Core Melt Ejection</u>

To cope with the possible effects of direct containment heating (DCH), the staff concludes ". . . that ALWR design should include a depressurization system and cavity design features to contain ejected core debris."

This is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is probably the preferred approach.

11. <u>Containment Performance</u>

The staff recommends that a containment performance guideline, expressed as a conditional containment-failure probability (CCFP) of 0.1, be used in evaluation of the ELWR designs. As an alternative, the staff proposes a deterministic performance goal that it believes would offer comparable protection.

We have previously recommended (ACRS Comments on An Implementation Plan For The Safety Goal Policy, dated May 13, 1987) such a quantitative guideline for containment performance as a part of the implementation of the Safety Goal Policy. However, this should be regarded as guidance to the NRC staff in its development of requirements for applicants. Merely passing on this guidance to applicants is not enough because the definition of CCFP is too imprecise. The deterministic performance criterion for containment systems suggested by the staff is also difficult to interpret.

We have undertaken an effort (ACRS report on Containment Design Criteria, dated March 15, 1989) to propose containment design criteria for future plants. But, as we said at the beginning of our study, we did not expect that it would directly affect the certification of the ELWR designs. This was, to some extent, because we recognized that our study would take some time to complete, but principally because the ELWR designs are now essentially complete and have been for some time.

We understand that the staff, assisted by the Brookhaven National Laboratory, is developing a regulatory guide that would serve as a basis for review of ELWR containment performance. We believe that the staff proposal will be adequate for ELWR review if it is supported by an appropriate regulatory guide developed on a timely schedule, and if it can be reasonably demonstrated that a containment that meets this guidance has a CCFP of not more than 0.1.

12. <u>ABWR Containment Vent Design</u>

During our April 5-7, 1990 meeting, we heard presentations from the staff and the General Electric Company regarding the staff's proposal that the Commission approve the use of severe accident design features that include a containment overpressure protection system in the ABWR design. We recommend that use of a containment overprotection system be approved subject to the results of the regulatory review.

13. Equipment Survivability

The staff recommends that features provided in the ELWR designs that are intended only for severe accident protection (prevention and mitigation) need not be subject to 10 CFR 50.49 (Environmental Qualification Requirements), 10 CFR 50, Appendix A (Redundancy and Diversity Requirements), and 10 CFR 50, Appendix B (Quality Assurance Requirements). However, the staff will require that mitigation features must be designed so there is "reasonable assurance" that they will perform their intended function in the severe accident environment and over the time span for which they are needed. Further, the staff proposes that at least one train of features provided for design basis accident protection, but also relied upon for severe accident protection, must be able to survive severe accident conditions for the time period that is needed to perform its intended function with "high confidence." In addition, the staff proposes to require that severe accident mitigation equipment be capable of being powered from an alternate power supply, as well as from the normal Class IE on-site systems.

To accomplish "reasonable assurance" and "high confidence," the staff will require that severe accident protective features use high quality industrial grade components which will be selected for the service intended and qualified by analysis or tests.

We endorse the staff's position. We note, however, that in this instance the staff's position includes much more than the underlined portions of the enclosure to SECY-90-016.

IV. NON-SEVERE ACCIDENT ISSUES

14. Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

The staff states that it has not yet developed a position on this issue that can be applied generically to all future designs and recommends that the Commission approve a design-specific approach. We have no objection to the staff considering exemptions to the requirement that the OBE be at least one-half the SSE, where this can be justified. We note that this has been done in the past for 14 plants at 9 sites, but in each case using site-specific data. Other bases for justification may have to be provided for un-sited standard plant designs.

In the longer term, we recommend that the staff and the industry attempt to develop a position that can be defined generically. One approach worthy of study would be to abandon the use of two earthquake levels for the design of structures, systems, and components. Instead, the design could be based only on the SSE, with appropriate load factors and limit states, and a smaller but more likely earthquake could be established as a threshold for plant shutdown and inspection.

15. <u>Inservice Testing of Pumps and Valves</u>

The staff proposes that certain aspects of the testing and inspection of pumps and valves be enhanced to ensure the necessary level of component operability for the ELWR designs. We endorse the staff's proposal with the following clarification and additions:

- Although not stated explicitly, we were told during the March 7, 1990 meeting of our Subcommittee on Mechanical Components that the staff intends to apply the requirements of Generic Letter 89-10 (Ref. 8) to the ELWR plants as well. We endorse this intention.

- We recommend that the staff's requirement for full-flow testing capability be extended to other safety-related valves (e.g., MOVs) not just check valves. The requirement for flow testing of MOVs is included in Generic Letter 89-10.

- We recommend that the staff resolve the issue of check valve testing and surveillance requirements for existing LWR plants and indicate how it is to be applied to the ELWRs prior to issuing the FDAs.

- We recommend that the staff be encouraged to entertain proposals from the FDA applicants regarding alternative ways of meeting the in-service testing and surveillance requirements.

Additional comments by ACRS Members Harold W. Lewis and James C. Carroll and ACRS Members William Kerr, David A. Ward, and James C. Carroll are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>Additional Comments by ACRS Members Harold W. Lewis and James C. Carroll</u>

Apart from one paragraph submerged as part of Item 1, this letter endorses the scattershot approach the staff has taken to the important question of regulation of new reactors. It therefore deserves to be called Camel II, in deference to the Committee's similar letter of January 15, 1987. The differences are that this list has in fact had more careful consideration, and that its elements originated with specific staff proposals. Indeed, in many cases the genealogy can be traced to industry initiatives, and the staff is simply proposing to make mandatory those things that the industry has previously proposed to do on its own. None of this pays the slightest attention to the Commission's Safety Goal Policy, nor is there any hint of an effort to seize this opportunity to move the regulatory process in the direction of coherence and consistency. This is a pudding without a theme.

Let us then try to provide some perspective, since the Committee has chosen not to do so.

The Committee has often commented on the central role of the Safety Goals in providing a focus and objective for the body of regulation. Since this list sets the tone for the licensing of the next generation of light-water reactors, it is particularly important that its relation to Commission policies, especially the Safety Goal Policy, be clear. At the risk of repetition, we, and we believe the Committee, have never urged that specific regulatory decisions (such as these) be judged individually in the context of the Safety Goals, but only that the body of regulation be judged in that light. Individual decisions must still be made deterministically, with expertise and good judgment, but as part of a coherent overall body of regulation. Still, fifteen items come close to being a "body", and it is informative to see the role of the Safety Goals in the formulation of the staff recommendations. The Safety Goal policy, and other commission policies, are supposed to provide the glue that binds the whole structure together.

In effect, the staff says that it has proposed to the Commission a set of new safety goals (SECY-89-102), the Commission has not acted on them, either way, and therefore the staff will use them as if the Commission had approved. While we sympathize with the staff predicament, we think that is entirely inappropriate. The staff proposals include such things as a core-damage probability of 1E-5 per reactor-year, a "large accident" probability of 1E-6 per reactor year (with a bizarre definition of large accident), and a so-called conditional containment-failure probability. Not one of these has been approved by the Commission, yet the staff has used them in formulating its proposed policies on these items. It has rationalized this usurpation of power by asking for Commission action on SECY-89-102, and by stating that its own safety goals are "consistent" with those of the Commission. Of course any set of goals more stringent than yours will be consistent with your own, and acceptance of this argument will mean that the staff can regulate beyond your policies, more or less at will. That is precisely the situation your original goals were intended to foreclose. The Committee has often recommended that your Safety Goals be used as a final statement of "how safe is safe enough", not as a rigid minimum level of safety, beyond which the sky is the limit. Of course the industry may well have good reason to go further, but that is another matter.

In addition, as your own OGC has pointed out in SECY-90-016, this has the potential to open a Pandora's box, in which each party to a licensing proceeding may be able to claim the rights the staff claims--to insist on improvements beyond the rules. You will have to face this problem at some time, and the sooner the better.

We do not wish to understate the difficulty involved in translating a safety-goal policy into a workable body of regulation. The Committee has written you of its own recommendations for an organized approach to that problem, but we believe it can and should do more. Nuclear safety is not helped by letting that problem fester--the fact that it is difficult is no excuse for inattention. It is too much to expect regulation to be coherent and rational in the absence of an objective for that regulation.

We do think it was useful for the Committee to respond to your specific request for technical help on the fifteen questions posed, but you should recognize that this was done in the absence of a measuring rod. Each item was therefore judged on its own, and the Committee has turned its back on the opportunity to respond in a structured and coherent way. Any one of these items might have come out differently if it had been measured against an underlying rationale. In our view, the Committee has forfeited a chance to be of real service to both you and the public.

<u>Additional Comments by ACRS Members William Kerr, David A. Ward, and James C. Carroll</u>

By the "rulemaking" approach to design certification the Commission has sidestepped the development of revisions to regulations that would reflect knowledge gained from experience and research over the last ten or more years. As a result, important new requirements are being imposed on applicants through a variety of staff actions and reactions. This is a loosely controlled process in which major policy decisions are made without an appropriate intensity of review. Contributing to the lack of discipline is what we believe to be a serious ambiguity in the Commission's policy on advanced reactors. The Commission has said it expects future reactors to be safer. But, whether this is a mandate or simply an expectation that a maturing industry will produce safer plants is not clear. The staff has interpreted it as a mandate and has translated this into an unauthorized extension of the safety goals. This is despite the statement in NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," published June 1988 (p. 4-1) that "the Commission expects but does not require enhanced safety margins other than those that may be required by the Safety Goal Policy." The Commission should not indefinitely postpone the development of a modern set of regulations. Only in this way will a proper balance be struck between adequate protection of the public health and safety and the advantages to the public that can come from efficient development of the nuclear power option.

<u>References</u>:

1. SECY-90-016, memorandum dated January 12, 1990, from J. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Evolutionary Light W[illegible]ctor (LWR) Certification Issues and Their Relati[illegible] Current Regulatory Requirements

2. Draft SECY-89-102, memorandum dated March [illegible], 1989, from V. Stello, Jr., Executive Director for Operations, NRC, to the Commissioners, Subject: Implementation of Safety Goal Policy

3. Electric Power Research Institute (EPRI), Advanced Light Water Reactor Requirements Document (Chapters 1 through 13), issued December 1987

4. General Electric Company, Advanced Boiling Water Reactor Licensing Review Basis Document, issued August 1987

5. U. S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," Volumes 1 and 2, dated June 1989

6. Memorandum dated March 8, 1990 from W. Minners, Office of Nuclear Regulatory Research, NRC, to R. Fraley, ACRS, Subject: Proposed Generic Letter on Individual Plant Examination for Severe Accident Vulnerabilities Due to External Events (IPEEE) and Supporting Documents (Predecisional)

7. Sandia National Laboratories, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NUREG/CR-5088, published January 1989

8. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," issued on June 28, 1989 to licensees for all power reactors, BWRs, PWRs, and vendors in addition to General Codes applicable to generic letters.

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

September 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: IMPLEMENTATION OF THE SAFETY GOAL POLICY

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we reviewed the Staff Requirements Memorandum (SRM) dated June 15, 1990, in regard to SECY-89-102 - Implementation of the Safety Goals.

As you are aware, the Committee has had considerable interest in this subject and has expressed its views in a number of reports to the Commission. We believe the Safety Goal Policy development has provided an opportunity for establishing a coherent philosophy of regulation and that it can be used to implement this philosophy. That has been the reason for our interest. Implementing the policy abstractions in a practical way will be a difficult undertaking, but one we believe to be achievable. While we have not reviewed this SRM in detail, we think it is an excellent first step. We applaud the efforts of the Commission in producing it, and look forward to future interactions as the policy implementation develops.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

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

April 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL

During its 360th meeting, April 5-7, 1990, the Advisory Committee on Reactor Safeguards reviewed the staff's proposed rule on nuclear plant license renewal. This matter was also discussed during a meeting of the Regulatory Policies and Practices Subcommittee on March 26, 1990. During these meetings we had the benefit of discussions with representatives of the NRC staff, NUMARC, EPRI, Northern States Power Company, and Yankee Atomic Electric Company. We also had the benefit of the referenced document and its enclosures.

The decisive issues for license renewal and associated plant aging, and the potential for further aging during the proposed license extension, should be addressed throughout the life of a plant. Attention to aging phenomena, and the criteria for safe operation (adequate protection of the health and safety of the public), should be the same just after as just before license renewal. There may be components or systems which are not aging issues during the first forty years, but become so later, and which therefore may require special attention.

At the time that the forty year period for a license was chosen, there was no special technical rationale for its choice, and no specific form of plant aging becomes magically decisive at forty. The regulatory job for license renewal is to identify the aging elements of the plant, and ensure that they receive timely attention during the extended license period.

In that context, we were surprised by the lack of emphasis on pressure vessel integrity during our briefings. This is surely one of the driving technical issues for extended life, and we assume that it will move to a more central position as the plans develop.

The staff proposes to use the "current licensing basis" of a plant as the basis for license renewal, but there seems to be some ambiguity about the interpretation of the term. The industry seems concerned that this may provide an opportunity to impose arbitrary new requirements. It is important that this terminology be clarified, so that any future conflicts of interpretation are minimized.

With these observations, we concur in the approach being proposed by the staff, which emphasizes attention to aging phenomena, avoids the temptation to treat license extensions as relicensing, and makes a timely start toward providing an integrated policy for dealing with aging phenomena.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
Memorandum dated March 6, 1990 from Warren Minners, Office of Nuclear Regulatory Research, NRC, to Raymond F. Fraley, ACRS, Subject: Proposed Rule on Nuclear Power Plant License Renewal, w/enclosures: Draft Commission Paper, "Proposed Rule on Nuclear Power Plant License Renewal"

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

June 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED RULE TO IMPLEMENT AN EMERGENCY RESPONSE DATA SYSTEM

During the 362nd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 1990, we discussed with members of the NRC staff their proposed rule to implement an Emergency Response Data System (ERDS). We also had the benefit of discussions with a representative of NUMARC and of the referenced documents.

In previous reports to the Commission dated May 6 and November 12, 1980, the Committee expressed its concern that a proposed nuclear data link could lead to inappropriate NRC involvement in the management of any future serious nuclear power plant accident. We continue to believe that all operational aspects of accident management must be the responsibility of the licensee/owner/operator. We are aware that the NRC has the authority and the responsibility to manage, for example, to give orders concerning the operation of a nuclear power plant, if that is determined by the Commission to be necessary to protect public health and safety. What concerns us is the possibility of informal intervention without the formal assumption of authority and responsibility, and in light of the examples of intervention during normal plant operation to be found in Appendix A of draft NUREG-1395 as referenced in SECY-90-80, "Draft Regulatory Impact Survey Report," our concerns are not alleviated by staff insistence that the existence of an ERDS will not make this more likely.

We recognize that the NRC has a responsibility to provide information to various governmental entities and to the media. However, we also looked for evidence, both in the documentation provided to us and in our meeting with the staff, that the existence of an ERDS would decrease public risk. Although both the documentation and the staff's presentation make statements that risk will be decreased, no evidence is provided to back up these assertions. Thus, if the ERDS is to be treated as a backfit, the benefit side of the ledger contains only statements that more information, and more accurate information, available in the NRC Incident Response Center, will decrease risk.

We were also told that the information that ERDS can provide would free the plant operational staff from the burden of having to respond to questions from NRC headquarters. However, since any serious accident is likely to result from and to produce unexpected phenomena, it is plausible to suppose that the existence of ERDS might actually increase the number of questions asked. We also observe that in station blackout, which many PRAs predict as a major risk contributor, ERDS would be unavailable.

We acknowledge that the system has positive aspects, but in our view they are outweighed by what we view as negatives. We therefore do not support the proposed ERDS, and thus we do not endorse the proposed rule.

If, however, it is decided to continue with ERDS, we recommend that it not be made compulsory until several years of experience have indicated the form it should take and the way in which it should be operated. Some of the problems that have occurred with the safety parameter display system could have been avoided if experience had been gained before these systems were mandated.

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

Sincerely,

Charles J. Wylie
Acting Chairman

Additional Comments by ACRS Members James C. Carroll, Ivan Catton, and Carlyle Michelson

Although we share the concerns about the accident management issue discussed in the Committee's letter, we believe that the potential benefits of the ERDS in supporting the NRC's role in accident situations are of sufficient importance to outweigh these concerns. Accordingly, we would recommend that the staff proceed with the proposed rulemaking unless it becomes clear that the present voluntary program will ensure industrywide participation.

References:

1. Draft Proposed Rulemaking Package (transmitted by note dated May 31, 1990 from Tony DiPalo, RES), with enclosures:
(1) Federal Register Notice of Proposed Rulemaking,

(2) Regulatory Analysis of the Proposed Rule Concerning the Emergency Response Data System, and
(3) NUREG-1394, "Emergency Response Data System (ERDS) Implementation"

2. U.S. Nuclear Regulatory Commission draft NUREG-1395, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities," dated February 1990.

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

August 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: LEVEL OF DETAIL REQUIRED FOR DESIGN CERTIFICATION UNDER PART 52

During the 364th meeting of the Advisory Committee on Reactor Safeguards, August 9-11, 1990, we reviewed the Commission Policy Issue Paper SECY-90-241 related to the level of detail required for design certification under 10 CFR Part 52. Our Subcommittee on Improved Light Water Reactors also reviewed this matter during a meeting on August 8, 1990. During these reviews, we had the benefit of discussions with representatives of the NRC staff and of NUMARC. We also had the benefit of the documents referenced.

Two important issues are addressed in SECY-90-241. The first deals with the level of detail to be included in an application for design certification under Part 52. The second deals with the level of detail to be included in the design certification rule itself. The first issue is of immediate importance and needs to be resolved before the NRC staff completes its review of the Standard Safety Analysis Report (SSAR) and other documents on which the application for design certification is to be based.

One might view the second issue as being less urgent, since it comes into play only after the application for design certification has been filed. At that point, one decides what portion of the information in the application is to be included in the design certification rule. However, we believe it is important for the staff to have an early awareness of the extent to which the information it is reviewing may become subject to revision during the design certification rulemaking. This would allow the staff to include appropriate wording in its Safety Evaluation Report (SER), identifying certain features for mandatory inclusion in the design certification rule. This would ensure that such features would not be changed in the future without the full protection of Part 52 design change requirements.

In SECY-90-241, the staff listed four options for the level of design detail that might be included in the application for certification and in the design certification rule. Unfortunately, they mixed the possible content of the application with the possible content of the rule. Only the Level 2 and Level 3 options appear to be open for serious consideration.

In the background statement for SECY-90-241, the staff points out that Part 52 is clear regarding the need for submittal of an "essentially complete design" when applying for design certification. The level of detail in a design certification application must be sufficient for the Commission to reach closure on all safety questions and establish assurances that future construction will be in conformance with the design. We believe the regulations are clear and proper concerning this required level of detail. The staff has indicated that both the Level 2 and Level 3 options will meet the requirements of Part 52.

From the viewpoint of what should be included in the design certification application, the Level 2 option stipulates that the depth of design detail submitted should be similar to that of a final safety analysis report for a recently licensed plant (minus site-specific and as-built information). In addition, the application is to contain information concerning features that ensure enhanced safety benefits from standardization. For the Level 3 option, the depth of design information submitted in the design certification application is less than that for Level 2 but still claimed to be sufficient for the staff to make its findings on all safety questions. We are not convinced that it is. We recommend that you adopt the Level 2 option because it ensures compliance with Part 52 requirements and the achievement of any benefits from that level of standardization.

Although we recommend that the level of detail submitted be that corresponding to the staff's Level 2, we do not believe that all of this information should be included in the design certification rule. We believe that some form of the two-tier approach proposed by NUMARC is essential from a practical point of view even though it may lead to some decrease in the degree of standardization.

Determining what goes into each of the tiers will require some trade-off between standardization and practicality and can have some effect on safety. We believe that the staff and the industry should be encouraged to develop criteria to define the division between the two tiers. As progress is made in this effort, we will review the proposed criteria and report on them to you if you wish.

Additional comments by ACRS Member Lawrence E. Minnick are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Member Lawrence E. Minnick

Neither the written material referenced above, nor our discussions with the staff has revealed any justification in terms of enhanced safety for standardization of plant designs beyond those portions directly and significantly related to safety.

Since it is clear that standardization, per se, is not an unmixed blessing, I strongly recommend that the ultimate degree of standardization should not be pursued for its own sake, but rather should be limited to that degree clearly essential to the assurance of plant safety.

Obviously competition among suppliers, and innovation and improvement in general, are considerably hampered by standardization. Those considerations have been so fundamental to this country's technical supremacy that they should require no elucidation here, but perhaps it does bear pointing out that standardization of nuclear units is inherently limited in any event, for example, by differing site characteristics and inevitable variations in operating experience.

I feel that the "two-tier" approach proposed by NUMARC will also alleviate the burden of standardization. I endorse that approach, which by reliance on the well-demonstrated 10 CFR 50.59 requirements will limit changes to those having no significant effect on safety.

References:

1. SECY-90-241, Memorandum dated July 11, 1990 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Level of Detail Required for Design Certification Under Part 52.
2. U.S. Nuclear Regulatory Commission, Rules and Regulations - 10 CFR Part 52, "Early Site Permits; And Combined Licenses for Nuclear Power Reactors," April 28, 1989

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

October 11, 1990

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

Dear Mr. Taylor:

SUBJECT: DRAFT IMPLEMENTATION DOCUMENTS FOR THE PROPOSED LICENSE RENEWAL RULE

During the 366th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 1990, we reviewed draft Regulatory Guide, Task DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," and associated draft NUREG-1299, "Standard Review Plan - License Renewal." Our Subcommittee on Plant License Renewal also reviewed this matter during its meeting on October 2, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. These documents are an important part of the program to implement the proposed license renewal rule, 10 CFR Part 54, that was published for public comment on July 17, 1990. We commented to the Commission on this proposed rule in our report of April 11, 1990.

We believe that the general approach proposed by the staff for implementation of the license renewal process is reasonable, and we agree that both of the subject documents should be published at this time for public comment. However, we have a concern, discussed below, about control of the process for selecting structures and components important to license renewals (SCITLRs). We believe that this matter should be considered further as public comments on the rulemaking are evaluated. We also offer several comments on the implementing documents.

There is justification for the general philosophy of the proposed license renewal rule. Aging-degradation issues should be dealt with by more explicit programs as the plant age passes beyond the general target age for which it was designed. Our understanding is that a 40-year operating life has been used for most structures and components in nuclear power plants. However, that target age and the design were not so precisely defined that there should be a step increase in licensing requirements as the plant passes its 40th anniversary of operation. As we said in our April 11, 1990 report, "no specific form of plant aging becomes magically decisive at forty." We have a concern that the license renewal process under the proposed 10 CFR Part 54 will permit or encourage a significant expansion of regulatory requirements as a plant phases into operation under a renewed license. We had hoped and expected that the implementing documents would provide some clear indications of how such regulatory expansion would be constrained. They do not. Introductory material in the proposed 10 CFR Part 54 indicated that the backfit rule would somehow be used in controlling the extent to which regulatory requirements would be expanded. However, the rule itself does not make it clear how this is to be done, nor do the draft implementing documents. We recommend that the rule or the implementing documents be revised to ensure that the process for selecting SCITLRs and developing new requirements is sufficiently disciplined.

In addition, we have several specific comments on the proposed implementing documents:

(1) In the proposed process for evaluating age-related degradation, the draft Regulatory Guide indicates that a decision about classification of a given structure or component should be made on the basis of whether the structure or component is routinely replaced or refurbished (see Block 12 of Figure 1B in the draft Regulatory Guide). We recommend that satisfactory results of inspection or monitoring should also be credited at this decision point.

(2) Many of the unresolved safety issues and generic safety issues that have been analyzed over the past several years have had assumptions about expected plant life factored into their resolution. The staff has indicated that, in general, an expected life of 60 years instead of 40 years would make little difference in cost-benefit analyses, given the large uncertainty inherent in the calculated results. However, the staff also indicated that a review of all such resolutions will be made, in the light of new expectations about plant lifetimes, given the changes of 10 CFR Part 54. We would like to be kept informed about the results of this review.

(3) Certain industry topical reports on the subject of aging degradation are being developed by NUMARC, and are expected to be approved by the staff as acceptable references in license renewal applications. We encourage the development of these industry reports as a means of providing a comprehensive technical base for license renewal reviews. Because the license renewal process can be expected to extend over many years, much technical information about aging will be in need of revision, and some means for formally updating these industry reports and their approval by the NRC should be provided.

(4) Perspectives gained from applicable risk assessment should be used in the selection of SCITLRs.

(5) Consideration should be given to including physical security systems in the SCITLR program.

We plan to continue our review of this important subject after public comments on this proposed rule, the Regulatory Guide, and the proposed Standard Review Plan are received and assimilated.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide, Task DG-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," Revision 5A dated August 1990, and U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Draft NUREG-1299, "Standard Review Plan, License Renewal," dated August 1990, transmitted by memorandum dated August 31, 1990, from Eric S. Beckjord, RES, and Thomas E. Murley, NRR, to Raymond F. Fraley, ACRS
2. U.S. Nuclear Regulatory Commission, Rules and Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Proposed Rule Making, Published July 17, 1990

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

December 10, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SECY-90-377, "REQUIREMENTS FOR DESIGN CERTIFICATION UNDER 10 CFR PART 52"

During the 368th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 1990, we reviewed the Commission Policy Issue Paper SECY-90-377 related to the requirements for design certification under 10 CFR Part 52. Our Subcommittee on Improved Light Water Reactors also reviewed this matter during a meeting on December 4, 1990. During these reviews, we had the benefit of discussions with representatives of the NRC staff and of NUMARC. We also had the benefit of the document referenced.

We commend the staff for its accomplishment in producing SECY-90-377 on a complex subject and in a relatively short time. In general, we concur with the staff's approach to design certification. We agree that the scope and level of detail should be similar to that required for a final safety analysis report (FSAR) at the operating license (OL) stage for a recently licensed plant (1985-90), without site-specific and as-built information. We concur with the graded approach of defining the level of design required, and the tiered approach proposed. However, we do not agree that the vast amount of information and level of detail that is proposed to be included with the application is needed for a safety determination. Therefore, we recommend that SECY-90-377 not be implemented as presently written.

SECY-90-377 appears to be driven by requirements for both standardization and safety. We recommend that the staff focus the scope on that needed for its safety determinations. In this regard, we propose that Tier 1 and Tier 2 information be limited to that required for the safety determination.

In general, we agree with the flexibility for making changes to the technical information. However, we believe that greater flexibility should be permitted for making changes to Tier 2 information following design certification. This flexibility would allow the necessary design refinements that are inevitable. We note that in

SECY-90-377 the staff proposes to provide for a process similar to that of 10 CFR 50.59 for making changes to Tier 2 information between Combined Operating License (COL) issuance and operation. We recommend that the same change process be permitted for the period beginning after design certification.

We recommend that the Commission instruct the staff to proceed with preparation of the proposed regulatory guide. The focus of the regulatory guide should be on that information required for the staff's safety determination.

We recommend that the Commission instruct the staff to update the Standard Review Plan so that it can support design certification reviews.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
SECY-90-377 dated November 8, 1990 from James M. Taylor, Executive Director for Operations, to NRC Commissioners, Subject: "Requirements for Design Certification Under 10 CFR Part 52"

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

February 15, 1990

The Honorable J. Danforth Quayle
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 has reported each year to the Congress on the Safety Research Program of the Nuclear Regulatory Commission.

In our December 18, 1986 letter to the Congress, we proposed to provide more focused reports on specific research issues rather than one all-inclusive annual report. The Commission agreed with our suggestion, and then NRC Chairman Zech submitted a legislative proposal in the form of a draft bill to the 100th Congress on December 2, 1987 to amend Section 29 of the Atomic Energy Act of 1954 to accomplish this. Since the 100th Congress did not act on this matter, he submitted a similar, but somewhat modified, legislative proposal on February 2, 1989 to the 101st Congress for consideration. We expect that the Congress will consider this matter during this year.

In the past year we have reviewed the NRC safety rsearch program and other closely related matters in the following areas:

° Accident Management Strategies

° Application of Leak-Before-Break Technology

° Containment Performance

° Containment Structural Integrity

° Embrittlement of Reactor Pressure Vessel Supports

° Fire Risk Scoping Study

° Human Factors Research Program Plan

- Inservice Inspection of Boiling Water Reactor Pressure Vessels
- Occupational Radiation Exposure to Skin from Hot Particles
- Piping Integrity
- Severe Accident Research Program Plan
- Thermal-Hydraulic Phenomena.

We have provided reports to the Commission on several of the matters mentioned above and copies of these reports are attached.

We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

*Attachments:

1. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: Additional Applications of Leak-Before-Break Technology, March 14, 1989
2. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: Proposed Severe Accident Research Program Plan, March 15, 1989
3. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: NRC's Human Factors Programs and Initiatives, May 9, 1989
4. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," May 9, 1989
5. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: Generic Letter Related to Occupational Radiation Exposure of Skin from Hot Particles, May 9, 1989

6. Report from David A. Ward, Acting ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: NRC Thermal-Hydraulic Research Program, June 15, 1989

7. Report from Forrest J. Remick, ACRS Chairman, to Kenneth M. Car , U.S. NRC Chairman, Subject: Proposed Staff Actions Regarding the Fire Risk Scoping Study (NUREG/CR-5088), July 18, 1989

8. Report from Forrest J. Remick, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Draft Supplement 2 to Generic Letter 88-20, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," November 20, 1989

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*For Items 1 through 8, see NUREG-1125, Volume 11, 4/90.

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

February 15, 1990

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 has reported each year to the Congress on the Safety Research Program of the Nuclear Regulatory Commission.

In our December 18, 1986 letter to the Congress, we proposed to provide more focused reports on specific research issues rather than one all-inclusive annual report. The Commission agreed with our suggestion, and then NRC Chairman Zech submitted a legislative proposal in the form of a draft bill to the 100th Congress on December 2, 1987 to amend Section 29 of the Atomic Energy Act of 1954 to accomplish this. Since the 100th Congress did not act on this matter, he submitted a similar, but somewhat modified, legislative proposal on February 2, 1989 to the 101st Congress for consideration. We expect that the Congress will consider this matter during this year.

In the past year we have reviewed the NRC safety research program and other closely related matters in the following areas:

° Accident Management Strategies

° Application of Leak-Before-Break Technology

° Containment Performance

° Containment Structural Integrity

° Embrittlement of Reactor Pressure Vessel Supports

° Fire Risk Scoping Study

° Human Factors Research Program Plan

° Inservice Inspection of Boiling Water Reactor Pressure Vessels

° Occupational Radiation Exposure to Skin from Hot Particles

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° Severe Accident Research Program Plan

° Thermal-Hydraulic Phenomena.

We have provided reports to the Commission on several of the matters mentioned above and copies of these reports are attached.

We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

*Attachments:

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2. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: Proposed Severe Accident Research Program Plan, March 15, 1989

3. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: NRC's Human Factors Programs and Initiatives, May 9, 1989

4. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, U.S. NRC Chairman, Subject: NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," May 9, 1989

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7. Report from Forrest J. Remick, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Proposed Staff Actions Regarding the Fire Risk Scoping Study (NUREG/CR-5088), July 18, 1989

8. Report from Forrest J. Remick, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Draft Supplement 2 to Generic Letter 88-20, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," November 20, 1989

*For Items 1 through 8, see NUREG-1125, Volume 11, 4/90.

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

April 11, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: NRC SAFETY RESEARCH PROGRAM BUDGET

During the 360th meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 1990, we discussed the proposed NRC Safety Research Program and budget for FY 1991. Our Subcommittee on the Safety Research Program met with the Executive Director for Operations, representatives from the Office of Nuclear Regulatory Research (RES), and the Office of Nuclear Reactor Regulation (NRR) on February 7, 1990, and discussed the proposed FY 1991 budget along with the rationale for the continually dwindling NRC Safety Research Program budget and the associated impacts. After considering the information gathered at these meetings, we find ourselves concerned, not so much about the proposed FY 1991 budget, but about the trend of continually diminishing funding for the NRC research program. Unless this trend is arrested, the overall effectiveness of the agency will be seriously compromised.

We have been critical of certain parts of the NRC research program in the past and remain so (Refs. 1-6). It is not our intent to address program deficiencies in this report, but to communicate our belief that a viable research program is an essential part of the NRC regulatory process. In the following paragraphs, we describe the reasons for our concerns about the research budget trend, and offer suggestions for change.

TREND IN THE RESEARCH PROGRAM BUDGET

Pertinent figures from the NRC budgets for fiscal years 1975, 1981, 1983, and 1991 follow:

Fiscal Year	Total Agency Funding (in constant 1975 dollars)	Total Agency FTEs	Research Program Support Funding* (in constant 1975 dollars)	No. of FTEs* for Research
1975	$148.1M	2006	$ 61.2M	94
1981	294.6M	3139	129.5M	155
1983	277.4M	3403	110.0M	140
1991	218.0M	3240	36.1M	120

When the total NRC budget increased markedly in the late 1970s and early 1980s, the research budget increased proportionally. However, since 1981 funding for research has been much more dramatically diminished than that for the agency. From 1983 to 1990, the research program support budget, in 1975 dollars, was reduced by a factor of three.

POSSIBLE EXPLANATIONS FOR THE RESEARCH BUDGET TREND

Among the reasons that might be offered for the trend in research funding are:

- The Commission has explicitly decided that research has become less important than other agency activities. It may have concluded that nuclear power has reached relative maturity and that most of the technical questions relating to reactor safety and regulation have been answered. In competition with other demands on resources (e.g., the belief that more inspections of operating plants are needed), research has taken a "back seat."

- Research funding has been reduced as part of a policy directed by the Administration or the Congress, perhaps for the reasons mentioned above.

- Given the government budgeting process, it is easier to reduce funding for NRC research, which is largely allocated to persons and institutions not on the NRC payroll, than to curtail or terminate regulatory activities that directly involve NRC employees.

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*Associated with actual research support which includes planning, coordination, and managing research projects. Does not include technical assistance support for developing rules and regulations, resolving generic and unresolved safety issues, or review of IPE/PRA submittals.

All of these reasons may have influenced the research funding trend, but we believe that the third reason has had a disproportionate influence. As evidence for this, staff presentations to us described the largest portion of the agency's budget, which includes funding for salaries, rent, travel, office accessories, etc., as "nondiscretionary." When pressed, the staff agreed that these funds were not really "nondiscretionary" in the sense that there is explicit guidance to that effect from the Commission.

HISTORICAL BENEFITS OF NRC RESEARCH

Since its inception, the NRC has expended over $2 billion (actual dollars) on research. Research has led to numerous important technical contributions to the NRC's regulatory program and nuclear safety. Several examples follow:

- In the thermal-hydraulics area, extensive research has confirmed that emergency core cooling systems would adequately respond to the worst credible loss-of-coolant accidents, resulting in revision to Appendix K, with a potential avoided capital cost of about $8 billion (Ref. 7). Later, improved methods of analysis provided guidance for responding to questions arising from the TMI-2 accident about plant operation, and have permitted optimizations in reactor systems and operations.

- Several elements of the plant aging research program have led the way in assessing the effects of aging on nuclear power plant components and structures. They have also led to the development of examination and testing techniques and the identification of the essential elements for managing the effects of aging. The results of these research elements constitute the principal technical basis for addressing the aging-related issues associated with nuclear plant life extension and license renewal.

- In the geophysics and seismic areas, NRC-sponsored research programs have provided better understanding of the Eastern U.S. seismicity, which has permitted more realistic assessment of risk from earthquakes.

- In the area of materials science, NRC-sponsored research has provided means to improve and ensure the reliability of inspection methods and has provided key information in managing problems of stress corrosion cracking in BWRs. Additionally, research has provided the means for dealing with the pressurized thermal shock issue. Other research has made it possible to improve reactor safety by justifying the elimination of unnecessary pipe supports.

- NRC-sponsored research has led the way in development of methods for risk analysis. In addition, research has made it possible for the NRC to come to grips with severe accident questions.

Beyond these technical accomplishments is another benefit which is not always explicitly recognized, yet is as important as the others. We believe it to be generally accepted that the NRC's research program has been an important contributor to the high technical quality of the staff. The research program has not only developed important safety information, but has attracted capable people to work for the NRC and its contractors, and has provided a resource of technical expertise to all activities of the agency.

REASONS FOR CONTINUING A COMPREHENSIVE RESEARCH PROGRAM

Important questions about nuclear safety and regulation remain unanswered. Applications of nuclear energy involve demanding technologies, and society expects nuclear activities to be carried out to extremely high standards of public and environmental safety. While analysis indicates that the NRC has been largely successful in its task of ensuring safe practices, significant uncertainties in risk predictions and lack of understanding of certain important phenomena remain. These involve technical areas such as components and materials performance, seismic risk, accident management, severe accident phenomena, and human behavior. Continuing research can gradually provide information and understanding that will be valuable in dealing with these questions and uncertainties.

In addition, it is necessary to maintain the technical quality and credibility of the NRC staff. We were told that the average age of the research staff is now about 50. Vital and consistently funded programs will retain the contributions of experienced researchers and attract capable new people to the agency, in both research and nonresearch positions.

Many of the manifestations of several years of decreasing research funding are already visible:

- Important research programs are being curtailed or terminated.

- The national laboratories are systematically moving their better people to more attractive programs.

- RES is having difficulty in attracting competent technical personnel with research experience, which has led to an overall reduction in quality.

- The results of several expensive experimental programs have been lost.

- University programs have essentially ceased to exist in most areas.
- The role of RES as a world leader in research has diminished.
- The use of large-scale and separate-effects facilities has ended.
- RES participation in major cooperative foreign experimental programs is diminishing.

CONCLUDING REMARKS

It is difficult to establish the proper magnitude of support for research. Two aspects should be considered.

First is the absolute magnitude. In 1975, NRC research was funded at $61 million. In 1981, research funding had increased to $197 million, which was about $130 million in 1975 dollars. In 1991, the budget calls for about $78 million for NRC research which is about $36 million in 1975 dollars. Appropriate funding for a research program must be sufficient to retain vitality in programs, personnel, and facilities. What is appropriate depends on a number of factors, many of them imponderables. The nature of important research questions, the existence or nonexistence of appropriate facilities, results of early research, and experience in plant operation are among them. In the face of these uncertainties, the Commission must make judgments about funding research. Our judgment is that the present research funding level is below the minimum. If there are further reductions, RES will not be able to support and maintain an effective research program.

The fraction of the total NRC budget allocated to research is also an important consideration. It is a measure of the extent to which research programs can be expected to help maintain the technical expertise of the agency. We mentioned above that the research budget has been reduced from over 40 percent in the earlier years of the agency to about 16 percent in 1991, and that may be further reduced by the Congress. We believe there is evidence that this is too low and suggest that a guideline of at least one-quarter of the agency budget is more appropriate for a viable research program.

Finally, we suggest that you not take just our word for it. The agency has in place an excellent panel of experts to advise the RES Director, namely the Nuclear Safety Research Review Committee. We suggest that they focus more on their primary mission, which is to advise on general safety research philosophy and long-range strategy, rather than on the details of specific ongoing research programs. They should consider questions of what might constitute a viable research program, in terms of the technical areas and funding requirements, both absolute and relative.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. ACRS Report dated March 15, 1989, from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: Proposed Severe Accident Research Program Plan.

2. ACRS Report dated July 7, 1988, from David A. Ward, Acting ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Research Related to Heat Transfer and Fluid Transport in Nuclear Power Plants.

3. ACRS Letter dated December 8, 1987, from William Kerr, ACRS Chairman, to Victor Stello, Jr., EDO, Subject: ACRS Comments on Memorandum from Victor Stello, Jr., EDO, dated October 7, 1987, Regarding the Embrittlement of Structural Steel.

4. ACRS Report dated September 11, 1987, from William Kerr, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: ACRS Comments on Code Scaling, Applicability and Uncertainty Methodology for Determination of Uncertainty Associated with the Use of Realistic ECCS Evaluation Models.

5. ACRS Letter dated July 15, 1987, from William Kerr, ACRS Chairman, to Victor Stello, Jr., EDO, Subject: ACRS Comments on the Embrittlement of Structural Steel.

6. ACRS Report dated July 15, 1987, from William Kerr, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: ACRS Comments on Draft NUREG-1150, "Reactor Risk Reference Document."

7. Letter dated February 8, 1985, from E. P. Rahe, Jr., Nuclear Safety Manager, Westinghouse Electric Corporation, to D. F. Ross, Office of Nuclear Regulatory Research, NRC, Subject: LOCA Margin Benefits.

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

September 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: REEVALUATION OF THE SALP PROGRAM

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we continued our deliberations on the SALP program. We were previously briefed during our 363rd meeting, July 12-13, 1990, by representatives of the staff concerning its reevaluation of the SALP program as described in SECY-90-189 dated May 25, 1990. We have also reviewed the staff requirements memorandum (SRM) dated August 10, 1990, related to SECY-90-189. In addition, we have reviewed the staff's Draft NUREG-1395, Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Plant Activities, dated March 1990, and the Survey of Staff Insights on Regulatory Impact (SECY-90-250) dated July 16, 1990. Finally, we discussed a letter dated September 4, 1990, that the Committee received from NUMARC on the subject of SALP and regulatory impact (copy attached).

In our letter to you dated December 21, 1989, which was based on a briefing from the NRC staff on the SALP program during our December 1989 meeting, we commented that this increasingly important element of the regulatory process was "out of control." We asked you to consider "... suspension of the program and issuance of no new SALP ratings until enough reform measures are instituted to lend credibility to the process." We recommended that you "... make a clear statement of the purpose of SALP ratings, insist that your staff implement that purpose and no other, insist that the staff not use the [SALP] ratings as weapons to enforce obedience to idiosyncratic policies that are not yours, greatly dilute the Regional autarchy in the process, and institute a workable set of checks and balances." (This latter point was further expanded in our letter of February 15, 1990, to you on the subject of Coherence in the Regulatory Process.) In your letter of February 2, 1990, you advised us that you planned no immediate action, as recommended by us, on the SALP program until the staff had completed its reevaluation.

On the basis of our review of the staff's reevaluation of the SALP program, as described in SECY-90-189 and as modified by the August 10, 1990 SRM, we have concluded that the recommended programmatic changes are appropriate and generally consistent with the objectives that have been defined for the program. However, we do not believe that these changes go far enough. We had expected that SECY-90-189 would address the issues raised in our letter of December 21, 1989, and this is clearly not the case.

The staff is planning to issue a paper in September on those changes in the regulatory program that it believes are suggested by the recent regulatory impact surveys. That would provide an excellent vehicle for the incorporation of changes designed to respond to the recommendations on the SALP program that we made in our letter to you. We urge you to make sure they do so.

We plan to meet with the staff in order to evaluate its proposed regulatory reforms including reforms to the SALP program that may go beyond SECY-90-189. We believe that such changes are needed in the interest of improving the overall coherence of the agency's regulatory process. This view appears to be strongly supported by the regulatory impact surveys of both licensees and staff members.

Additional comments by ACRS member Carlyle Michelson are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Attachment:
Letter dated September 4, 1990 from Joe F. Colvin, NUMARC, to Harold W. Lewis, ACRS, w/attachments

Additional Remarks by ACRS Member Carlyle Michelson

It is my view that the staff's reevaluation of the SALP program, as described in SECY-90-189 and as modified by the Commission SRM, adequately addresses the SALP program issue. Thus far, it is not clear that any other changes in the program are needed. I agree that the staff should be instructed to respond to our recommendations on the SALP program in its planned September 1990 SECY paper.

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

1776 Eye Street, N.W. • Suite 300 • Washington, DC 20006-2496
(202) 872-1280

Joe F. Colvin
Executive Vice President &
Chief Operating Officer

September 4, 1990

Dr. Harold W. Lewis
Chairman, Subcommittee on Regulatory Policies and Practices
Advisory Committee of Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Lewis:

In reviewing the agenda for the next ACRS full committee meetings of September 6-8, 1990, we became aware that the Systematic Assessment of Licensee Performance (SALP) changes recently proposed by the NRC staff would be the subject of an ACRS report to the Commissioners. The purpose of this letter is to make you aware of industry concerns in this area that we are discussing with the NRC commissioners and senior staff. Specifically, we are concerned that the SALP process is being decoupled from the overall issue of regulatory impact and that changes are being made that affect the overall regulatory process without the root causes of the problems described by both the industry and the staff having been properly identified.

The SALP process has a significant impact on licensee activities and is a major cause of the problems identified by the recent Regulatory Impact Survey. The NRC staff's assessment of industry feedback, as contained in draft NUREG-1395, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities," identified problems in the SALP process as one of the two principle themes emerging from all licensees' comments. Specifically, the report concluded that "licensees acquiesce to NRC requests to avoid poor numerical Systematic Assessment of Licensee Performance (SALP) ratings and the consequent financial and public perception problems that result, even if the requests require the expenditure of significant licensee resources on matters of marginal safety significance;". Further, the recently released "Survey of The NRC Staff Insights On Regulatory Impact," SECY-90-250, confirmed the findings of draft NUREG-1395, stating that "...licensees are extremely sensitive to NRC activities and sometimes acquiesce to avoid confrontation that could create the perception that they are unresponsive. This makes licensees vulnerable to potential abuses of regulatory authority."

On May 14, 1990, we wrote to Chairman Carr (copy attached) commending the efforts of the NRC and staff on the draft NUREG-1395 and stressing the need to evaluate all the information available, determine the root cause, and develop a plan and schedule to make corrections to the process. Further, we offered the industry's assistance to help achieve our mutual goal of improvements to the regulatory environment. Chairman Carr responded in a letter (copy attached) of June 21, 1990, indicating that the plan and schedule developed by the staff will be sent to the Commission, the ACRS, and be made publicly available, and that industry views on the plan and schedule would be welcome at that time.

In our view, plans and recommendations related to the issue of regulatory impact, including future changes to the SALP, should be subject to industry, as well as public, review and comment before action is taken by the Commission. In that manner the Commission will have the comments of all interested parties as input to their decision-making process.

We would be pleased to discuss this matter further with the ACRS.

Sincerely,

Joe F. Colvin

Joe F. Colvin

JFC:ben
Attachments

cc: Mr. Carlyle Michelson
Mr. Charles J. Wylie

NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

1776 Eye Street, N.W. • Suite 300 • Washington, DC 20006-2496
(202) 872-1280

Byron Lee, Jr.
President & Chief
Executive Officer

May 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. - 20555

Dear Chairman Carr:

We commend the effort of the NRC and Staff to assess the impact of NRC activities on utilities through the Regulatory Impact Survey. This is an important step which can lead to an improved interface between the regulator and the regulated industry and, thus, towards a greater margin of safety.

The survey addresses many of the concerns about the regulatory burden and uncertainty expressed by the industry over the past several years. We believe the draft survey report, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities" (draft NUREG-1395), contains an excellent summary by the Staff of the candid comments provided by the personnel from various levels within the licensees that participated in the survey. We commend the Survey Team for an excellent job of listening and reporting the information received. The licensee input represents a sincere response to your challenge to U.S. utilities to give you specifics, a challenge you reinforced at the INPO Chief Executive Officer Conference last Fall.

We also believe the Staff's efforts to understand this information, both fact and perception, is vital to your efforts to improve the process. We are concerned with the statement in the Preface that, "In some cases, the perceptions and opinions given are at variance with the staff's understanding of the facts." We are not sure how to interpret this statement, but hasten to add the survey comments came from enough levels within each company and from virtually all companies to be more than perceptions. Also, they came from the people actually impacted. If real benefit is to be gained from this effort, the staff should apply the same principles they ask the licensees to apply: Evaluate all the information available, determine the root cause, and develop a plan with an implementation schedule to make corrections to the process consistent with your regulatory responsibilities.

We are anxious to assist the Commission to interpret the information received. We plan to coordinate further industry activities on these matters in order to minimize the burden. Please contact me or Joe Colvin as to how we may be of further assistance.

Sincerely,

Byron Lee, Jr.

Byron Lee, Jr.

BL:exc

cc: Commissioner Thomas M. Roberts
Commissioner Kenneth C. Rogers
Commissioner James R. Curtiss
Commissioner Forrest J. Remick
Mr. James M. Taylor

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

CHAIRMAN

June 21, 1990

Mr. Byron Lee, Jr.
President & Chief Executive Officer
Nuclear Management and Resources Council
1776 Eye Street, N.W., Suite 300
Washington, D.C. 20006-2496

Dear Mr. Lee:

I am responding to your letter of May 14, 1990, concerning draft NUREG-1395, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities." I agree that the Regulatory Impact Survey (RIS) is important to the staff's efforts to improve regulatory activities, and I appreciate your willingness to assist the NRC in interpreting the information collected to date. However, I believe that any additional comments and suggestions that NUMARC may want to contribute to this effort would be more useful to the Commission at a later point in time.

As you may know, we have more to do on the overall program to assess regulatory impact. The Regulatory Impact Survey includes two other activities to solicit information. One activity consists of a questionnaire to all nuclear utilities soliciting voluntary information concerning management time devoted to all inspections and audits. The other activity is an internal survey of NRC staff on its perceptions of the impact that NRC licensing and inspection activities have on nuclear plant operation. A comprehensive evaluation of licensees' comments by the NRC staff is ensured by the inclusion of these two activities in the RIS program.

Following completion of the surveys, senior NRC managers will evaluate carefully all of the information obtained and will then develop a plan and implementation schedule to make corrections to the regulatory process consistent with our regulatory responsibilities. The plan and schedule will be forwarded to the Commission, made available to the ACRS in their role as advisors to the Commission, and made publicly available. The Commission believes that your views would be most helpful if they are focused on the plan and schedule, and we would welcome any additional comments and suggestions that you may want to make at that time.

Sincerely,

Kenneth M. Carr

Kenneth M. Carr

RECEIVED JUN 2 5 1990

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U.S. NUCLEAR REGULATORY COMMISSION

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This compilation contains 31 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1990. 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 divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

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