United States Nuclear Regulatory Commission
Office of Public Affairs
Washington, DC 20555
Phone 301-415-8200   Fax 301-415-2234
Internet:opa@nrc.gov

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(Tuesday, November 26, 1996)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received the four attached reports from its Advisory Committee on Reactor Safeguards. The reports, in the form of letters, provide comments on:

-- Plant-specific application of safety goals.

-- Position on direction setting Issue 32 - Future role of NRC research.

-- Proposed rule on steam generator integrity.

-- NRC programs for risk-based analysis of reactor operating experience.

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Attachments:
As stated

November 18, 1996

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

Dear Chairman Jackson:

SUBJECT: PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we discussed the application of Safety Goals on a plant-specific basis. This subject was also discussed at meetings of our Joint Subcommittees on Probabilistic Risk Assessment and Plant Operations on July 17-18, 1996, and of our Subcommittee on Probabilistic Risk Assessment on August 7, 1996. We also had the benefit of the documents referenced.

In a Staff Requirements Memorandum dated June 11, 1996, we were requested to provide recommendations on how the Commission's Safety Goals and Safety Goal Policy should be revised to make them acceptable for use on a plant-specific basis.

The Safety Goal Policy Statement made it clear that the Quantitative Health Objectives (QHOs) and the subsidiary Core Damage Frequency (CDF) goal were to provide standards for the NRC staff to judge the overall effectiveness of the regulatory system. That is, if the risk posed by the population of plants on the average proved to be less than the Safety Goals, then the staff (and presumably the public) would deem that the regulatory system had functioned appropriately to protect the health and safety of the public.

The Safety Goals quantified "how safe is safe enough" for the population of U. S. plants. For an individual plant, however, the acceptable level of risk is determined by the concept of "adequate protection," which in the final analysis means compliance with the body of regulations. Risk-informed analyses would provide a more rational basis for making regulatory decisions regarding plant-specific requests for exemptions from the rules or for changes to the licensing basis, and the acceptability of new regulations.

In our August 15, 1996 report, we stated: "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria based on core damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met."

In developing plant-specific criteria, it is important to consider the regulatory needs in the near future and to ensure that the process will be evolutionary rather than so revolutionary that it might discourage the licensees from using this approach. It appears that most of the anticipated licensee requests for changes to their current licensing basis will deal with Level 1 probabilistic risk assessment (PRA) issues, e.g., inservice inspection, extension of allowed outage times. Furthermore, most licensees have only recently familiarized themselves with Level 1 PRA methodology for the narrow regime of power operations. They are just beginning to integrate findings of such Level 1 risk assessments with the safe operation of their plants. Even the NRC staff is still coming to grips with the implications of Level 1 risk assessment results for regulation of nuclear plants. Many licensees do not have access to the technologies for facile conduct of full-scope Level 2 or Level 3 PRAs that treat power operations, low power/shutdown operations, as well as accidents initiated by external events. Commonly accepted standards for such extensive, in-depth analyses do not exist.

An evolutionary and pragmatic approach for using Safety Goals on a plant-specific basis would be to use the CDF as the primary criterion for evaluating proposed changes along with a qualitative or quantitative evaluation of the possible Level 2 and Level 3 PRA issues raised by these changes. For a quantitative analysis, the following two options are offered:

1) Full-scope Level 2 PRA (with fission product transport capability).

To use this option, a conservative value for a LERF criterion must be determined. This value, along with the CDF criterion, will provide an acceptable basis for decisionmaking. We note that both the NRC staff and the Electric Power Research Institute, in its, "PSA Application Guide," are proposing the use of LERF as an acceptance criterion.

2) Full-scope Level 2 PRA (without fission product transport capability).

To use this option, conservative values for early containment failure frequency criteria for different reactor designs must be determined. These values, along with the CDF criterion, will provide an acceptable basis for decisionmaking.

In the longer term, we believe the agency should move beyond the evaluation of risk associated with proposed changes to individual plant licenses and apply the Safety Goals to assess the acceptability of plant-specific risk. This could be done in terms of the QHOs, along with the CDF, or in terms of the CDF and LERF. To use the QHOs directly, it would be necessary to have full-scope Level 3 PRAs. We believe that the use of Level 3 PRAs in the future should be encouraged.

Sincerely,

/s/

T. S. Kress
Chairman, ACRS

References:
1. Staff Requirements Memorandum dated June 11, 1996, from John Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Meeting with ACRS, Friday, May 24, 1996
2. ACRS report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters
3. Electric Power Research Institute Report TR-105396, "PSA Application Guide," prepared by ERIN Engineering and Research, Inc., August 1995

November 19, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Jackson:

SUBJECT: POSITION ON DIRECTION SETTING ISSUE 22 -- FUTURE ROLE OF NRC RESEARCH

During the 435th and 436th meetings of the Advisory Committee on Reactor Safeguards, October 9-12 and November 7-9, 1996, respectively, we reviewed Direction Setting Issue (DSI) 22. At the 435th meeting, we discussed this issue with the NRC staff. We also had the benefit of the documents referenced.

Direction Setting Issue 22 raises the question of what role the Office of Nuclear Regulatory Research (RES) will have in the future. A range of possible roles is defined in the discussion. These vary from elimination of a research capability at NRC to continuation of the research at its current, diminished level on a broad range of topics. The preliminary thinking is to select the continuing "business as usual" role for RES.

We contend that, first, changes are occurring within both the nuclear industry and the regulatory community that make it essential for NRC to have a research function. Second, we contend that a "business as usual" approach to NRC research is too timid. There is an urgency for the NRC to have research information to meet its obligations to protect the public health and safety in a changing environment. Finally, we contend that the planning for future research should be directed toward areas of focused need. In particular, research is needed to support NRC's transition to risk-informed and performance-based regulation.

The research arm of NRC has occupied a central role in the development of the body of regulations needed to ensure public health and safety in the commercial use of nuclear power. Since the division of the Atomic Energy Commission into the NRC and what eventually became the Department of Energy, RES has overseen the work needed to develop the design-basis analysis of nuclear power plants. This has included ensuring through a combination of experimental and analytical research that the analyses done for Appendix K to 10 CFR Part 50 are on a sound technical foundation. RES has also undertaken a vast effort to understand the residual risk posed by the use of nuclear power through the studies of severe accidents and the associated radionuclide source terms. RES has, in fact, been responsible for the evolution in the analysis of reactor safety from the bounding and the qualitative to the use of quantitative risk analysis.

From the pinnacle following the accident at Three Mile Island, RES has suffered a continuing scale-back of the activities it can afford to undertake. As with many institutions facing budgetary pressures, the longer term benefits of research activities have been sacrificed to ensure that there is the necessary financial backing for day-to-day activities that are the responsibility of NRC. NRC's research budget has, then, suffered disproportionately when funding cutbacks have been inflicted on NRC as a whole. Today, the available funding for research is, indeed, small enough that it is a legitimate question whether a viable research program can be maintained.

At the time these cutbacks in research funding have been taking place, changes have also been taking place in the way society deals with safety regulation. Most directly obvious has been the effort supported by both the Executive Branch and by Congress to base regulation on actual risk rather than bounding conservatism. The Vice President heads a Government-wide effort to base regulation, including regulation of nuclear power, on risk. Relative to most other regulatory agencies, NRC is well on the way to developing a risk-informed and performance-based regulatory system. NRC may well set an example for other regulatory agencies in this regard. It is, then, important that this be a good example.

A second societal development that will have safety implications is the economic deregulation of electrical power generation. This development has yet to be fully realized, but already efforts are being undertaken by the nuclear utilities to achieve greater economic competitiveness. Increases in reactor operating power and the extension of fuel life are just two immediate steps the industry is taking that have obvious safety implications. It is widely forecast that draconian measures will be necessary in the future to maintain nuclear power as a viable option for the generation of electrical energy. There are, of course, other changes taking place in the industry that fall in the domain of NRC such as plant aging; plant decommissioning; development of new, passive plant designs; and disposal of nuclear waste.

NRC is making great efforts to respond to the challenges posed by societal and industrial changes that are now taking place. The information available to the agency to meet these challenges is, however, proving to be limiting. By way of examples, consider the following:

- NRC is attempting to develop a risk-informed and performance-based regulatory system to improve the safety of nuclear plants and to relieve the industry of unnecessary burden.

But, NRC is trying to do this without any detailed knowledge of shutdown risk because RES is unable to fund studies comparable to the NUREG-1150 studies performed to understand risk during power operations.

- NRC development of probabilistic methods has not kept pace with its needs. Methods to treat human errors of commission or the impacts of organizational factors and management practices on risk are not available. Experience shows that human errors, organization, and management are responsible for or contributing to many accidents and "near misses."

- NRC wants to regulate in light of risk, but there is now only the technical capability for routine, noncontroversial evaluation of core-damage frequency. The capacity to extend estimates of core-damage frequency to evaluate risk has not been made widely available. There is not even consensus on how accurately analyses of risk, given that core damage has occurred, must be done nor how comprehensive such analyses must be.

- The introduction of digital instrumentation and control (I&C) systems in nuclear power plant safety related systems requires NRC to have the capability to regulate high-reliability software-based systems. NRC's understanding is limited to current software engineering methods which employ highly disciplined development process to design and produce high-reliability software. A consequence of this approach is lack of well-developed methods for evaluating the product of the process. NRC is limited to regulating the process of design and development of digital I&C systems because no accepted tools are available for evaluating the product.

- Financial constraints forced NRC to allow its codes for predicting fuel behavior to atrophy so they are no longer up to the state of the art. These codes cannot adequately predict fuel and clad behavior at burnups now being used by licensees. Recovery actions by RES have been constrained by resource limitations to narrow topical areas.

- NRC has not yet been able to formulate a risk-informed and performance-based fire protection rule to replace Appendix R to 10 CFR Part 50 which has been the source of so many exemptions and other controversies.

- NRC's opportunities to leverage its research budget by participating in international research consortia are becoming limited as NRC has less to contribute to the consortia efforts.

- NRC finds it must evaluate new, passive plant designs using tools developed for older plant designs because it cannot afford to develop analytical tools better suited for the simulation of the physics of these new designs. NRC must "make do" with computational tools that are now over a decade old and don't even begin to take advantage of all the more recent advances in computer technology.

- NRC is unable to predict or detect newly discovered modes of degradation of the primary pressure boundaries of pressurized water reactors. It has been forced to use rules designed to deal with wastage and corrosion of steam generator tubes to protect against a variety of forms of stress corrosion cracking.

There is clearly a need for a more aggressive NRC research program to confront the many challenges that the agency continues to face. We can be confident that the agency will meet its obligation to protect the public health and safety. But, without up-to-date tools produced by a forward thinking research organization, the agency will have to resort to methods that do not contribute to either regulatory efficiency or economic efficiency of the nuclear industry.

The financial resources now available to the agency for performing research are indeed limited. It has been necessary to make hard choices on what is to be done and what must be abandoned. A significant factor in the thinking on what is to be supported and what is to be abandoned has been a desire to preserve technical capability. This effort to preserve technical capability appears to have:

- led to an emphasis on the things that the agency knows best such as the thermal hydraulics of existing reactors, and

- diluted the efforts in many areas to preserve the current organizational units of RES.

The preliminary decision for DSI 22, which is to continue conducting research as it has been done in recent years, appears to enforce this emphasis on what is known well and to preserve the existing organizational structure of RES.

It is our position that more aggressive options need to be developed in response to DSI 22. One of these options is to focus the research in areas to meet the agency needs as it embarks on its experiment with risk-informed and performance-based regulation. The goal of research, then, ought to be, first, to provide risk information that is far more comprehensive than that now available, and then, to identify the performance indicators that do indeed reflect the risk. Furthermore, efforts are needed to use plant data and event reports to assess the adequacy of current probabilistic risk assessment methods.

RES also needs to anticipate safety implications that licensees will make in response to economic pressures. RES should be in a position to provide tools suitable for the safety evaluation of these changes. To do this, the split of work by RES between user requests and self-directed work may have to be reevaluated.

Sincerely,

/s/

T. S. Kress
ChairmaN, ACRS

References:
1. United States Nuclear Regulatory Commission, "Strategic Assessment and Rebaselining Initiative, Stakeholder Involvement Process Paper," dated September 16, 1996
2. United States Nuclear Regulatory Commission, "Strategic Planning Framework," dated September 16, 1996
3. United States Nuclear Regulatory Commission, "Strategic Assessment Issue Paper," dated September 16, 1996

November 20, 1996

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE ON STEAM GENERATOR INTEGRITY

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we reviewed the technical bases for the proposed steam generator integrity rule and an associated regulatory guide. During the 432nd meeting of the ACRS, June 12-14, 1996, and meetings of the Joint Subcommittees on Materials & Metallurgy and on Severe Accidents, June 3-4 and November 5-6, 1996, we heard presentations on subjects related to this matter. During these reviews, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute, and the Electric Power Research Institute, as well as the author of a differing professional opinion. We also had the benefit of the documents referenced.

The proposed steam generator integrity rule is intended to provide a risk-informed and performance-based regulation to replace an existing prescriptive regulation. In its present form, the rule is a performance-based regulation almost completely divorced from any direct relation to risk objectives. Such a performance-based rule proliferates the incoherence problems of the present deterministic approach. The proposed rule preserves a tenuous connection between "design-basis space" and "risk space" without clearly articulating the risk objectives.

Some of the characteristics exhibited in the development process of the rule and regulatory guide include the following:

- difficulty in reaching agreement on the performance criteria,

- incomplete and sometimes perfunctory analyses required to provide an assessment of relative risk,

- reliance on core-damage frequency alone as an indicator of risk, and

- recourse to defense-in-depth without specific criteria for its use.

We believe that more direct consideration of risk could have avoided some of these difficulties.

A controversial element of the proposed rule and regulatory guide is the introduction of severe accident issues into an area that has been exclusively resolved by using a design-basis analysis. This extension of the scope of accident analysis is necessary to make risk-informed regulatory decisions and is part of the cost of moving toward risk-informed regulation. Since licensees have done risk-informed analyses for the Individual Plant Examination (IPE) process, we believe that the analysis for addressing severe accident events should not be overly burdensome to them.

Steam generator tube ruptures are small contributors to the total core-damage frequency, but may be risk significant due to containment bypass effects. In previous analyses, the staff performed limited assessments of primary side fission product attenuation and neglected secondary side attenuation. The regulatory guide now proposes that the licensees deal with the risk of a thermally induced tube failure either by demonstrating that the frequency of the initiating events is sufficiently low (10^{-6}/reactor year) or by demonstrating that the conditional probability of tube failure, given that an initiating event has occurred, is low (on the order of 0.1). We believe that licensees should also be given the option to demonstrate that, even if thermally induced tube ruptures occur, the associated risk is low when a more realistic treatment of fission product attenuation is made.

We are concerned that the proposed regulatory guide, as presented, could send the wrong message to licensees that risk-informed and performance-based requirements are add-ons to the traditional design-basis accident approach and can only result in an additional burden. We believe that to be risk informed and performance based, the regulatory guide should begin with a clear statement of its objectives, followed by a statement of the performance criteria and the guidelines for meeting the criteria. We note that the staff has stated that the proposed performance criteria have been derived from risk analyses, but we have not seen these analyses. Rewriting the regulatory guide is not a trivial task, but could result in a regulatory framework that could be used as a model for future risk-informed and performance-based rulemaking efforts.

In other applications of performance-based regulation such as the Maintenance Rule, the licensees have been permitted to determine appropriate performance criteria and have been given more flexibility in developing the methodology used to determine whether the criteria have been met. For the steam generator rule, the staff has concluded that it should approve the performance criteria that are proposed by licensees to implement the steam generator rule. We agree with the decision of the staff that it should approve the criteria. Industry, however, should be provided more flexibility to propose alternative performance criteria supported by an appropriate risk analysis. We would like to review all of the supporting documentation before commenting on the specific criteria that have been proposed in the regulatory guide.

The demonstration that the criteria have actually been satisfied requires a complex process of nondestructive examination and evaluation of structural integrity and leakage during operation and design-basis accidents. The methodology required for these evaluations is not well established. Thus, the staff has felt constrained to provide a great deal of detail in the proposed regulatory guide to describe the characteristics of an acceptable methodology. Although we are not yet prepared to endorse the regulatory guide, we believe that the present immaturity of the methodology and the importance of the results justify such an approach.

The staff position is that the regulatory guide provides sufficient guidance for developing an acceptable methodology and that formal review of industry-developed repair criteria and procedures will not be required. We would like to review the results of a "trade study" of the preapproval approach vs. the post-implementation inspection approach to methodology acceptance.

Industry has questioned whether safety factors proposed in the steam generator rule are more conservative than those required by the ASME code. We encourage the staff to consider the industry's arguments.

Industry accepts the performance criterion proposed by the staff for primary-to-secondary leakage. Industry stated that this leakage criterion ought not be *ipso facto* a trigger for inspection or enforcement of regulations concerning the steam generator rule. This is a valid concern. Excessive leakage does not necessarily indicate a failure of the steam generator program. Adequate opportunities for staff action are available if failures of the program are discovered following a plant shutdown due to excessive primary-to-secondary leakage.

We are looking forward to reviewing the staff NUREG report concerning the staff's treatment of thermally induced tube failure. We are especially interested in the treatment of elevated temperatures resulting from flow through leaking tubes, and coupling between aerosol deposition and thermal hydraulics.

A differing professional opinion (DPO) was filed on July 11, 1994. We have reviewed the contentions in that DPO and summarized them in the attachment. We also note that Generic Safety Issue (GSI)-163, "Multiple Steam Generator Tube Leakage," identified in 1992 has yet to be prioritized and resolved. Both the DPO and the GSI are directly related to the proposed rulemaking. We urge the staff to prepare a point-by-point response to the issues in the DPO and to prioritize and resolve GSI-163 before implementing the steam generator integrity rule.

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

Sincerely,

/s/

T. S. Kress
Chairman, ACRS

Attachment:
Summary of Differing Professional Opinion Issues - Presented to the ACRS on November 7, 1996

References:
1. Memorandum dated October 25, 1996, from Brian Sheron, Office of Nuclear Reactor Regulation, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of the Proposed Steam Generator Rule [forwarding the proposed steam generator rule and draft steam generator regulatory guide]
2. Memorandum dated May 1, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, Subject: Resolution of Differing Professional Opinion Regarding Voltage-Based Repair Criteria for Steam Generator Tubes, dated July 13, 1994
3. Memorandum dated July 15, 1994, from James M. Taylor, Executive Director for Operations, NRC, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review Of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes [forwarding Differing Professional Opinion]
4. Report dated September 12, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
5. Memorandum dated September 30, 1994, from Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Comments On ACRS Review Of Generic Letter "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes"

SUMMARY OF DIFFERING PROFESSIONAL OPINION ISSUES
PRESENTED TO THE ACRS ON NOVEMBER 7, 1996

The DPO author estimates core-damage frequency with containment bypass to be 1 E-4 to 3.4 E-4 events/year. He stated that the uncertainties associated with characterizing steam generator tube defects and severe accident phenomena are not sufficiently understood to properly model tube rupture events. Tubes may fail before the surge line due to:

- crack networking and characterization of flaws not being adequately determined by nondestructive examinations,
- increased heat transfer caused by flow through tube cracks,
- cracks in tubes opening due to increased pressure,
- cracks in tubes unplugging at elevated pressure, and
- jets from tube cracks eroding adjacent tubes.

The DPO author stated that the staff should document the assumptions and models used to study hidden uncertainties.

November 22, 1996

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

Dear Mr. Taylor:

SUBJECT: NRC PROGRAMS FOR RISK-BASED ANALYSIS OF REACTOR OPERATING EXPERIENCE

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we reviewed the NRC programs for risk-based analysis of reactor operating experience. We heard presentations by and held discussions with representatives of the NRC staff regarding programs of the Office for Analysis and Evaluation of Operational Data (AEOD) including system reliability studies, risk-based performance indicators (PIs), accident sequence precursor (ASP) studies, and common-cause failures (CCFs). In addition, our joint Subcommittees on Probabilistic Risk Assessment (PRA) and on Plant Operations met with representatives of the NRC staff and its contractors on July 17 and October 30, 1996, to review these matters. We also had the benefit of the documents referenced.

The AEOD staff presented a summary report of its programs for risk-based analysis of reactor operating experience. We found these programs to be comprehensive in covering the collection and analysis of operational safety data based on operating plant experience and balanced in providing results to both the immediate assessments for the NRC's plant PIs and the continuing longer range assembly of useful databases for system performance including CCF rates. We are convinced that careful review of operating experience is the most applicable source of information that the NRC and the industry have to validate system reliability analysis models and predictions, and is the best source of data for future use.

These databases have been developed through significant resource expenditures by the industry and the NRC. Both share the results of this effort through their independent analyses of event reports, system reliability data, etc. This information can be made useful only if the results are carefully reviewed for insights into system reliability, human performance, and utility and NRC management practices that may affect safety. The AEOD programs reflect an awareness of the need to analyze these data intensively; however, the resources to perform a full scope analysis are not currently available. We urge that the priority assigned to this effort be revisited.

The NRC and the Institute of Nuclear Power Operations (INPO) have worked very hard to negotiate a more extensive sharing of their individual analysis products.

These efforts have had some success, namely, NRC has gained access to data in the Nuclear Plant Reliability Data System of INPO, thus expanding the bases for NRC compilation of CCF data. Some concerns remain with regard to the protection of INPO proprietary rights. We believe any database used by NRC on CCF should be accessible to the public.

The CCF database that has been developed is a significant technical step forward. AEOD uses the database for generic evaluations. Plant-specific evaluation will almost certainly require modification to reflect configuration differences between the specific plant being considered and AEOD's generic evaluations. Provision should be made to caution any users of the CCF database of the limited applicability in its current form and, if possible, provide guidance on the proper process for modifying the database to reflect specific plant characteristics.

The AEOD staff presented some information on planned revisions to the NRC's PIs and initial efforts to incorporate risk-based PIs into the program. We look forward to further examination of candidate indicators. They must be carefully selected with a clear understanding of how the connection to risk is made and how this connection can be quantified. A first step will be the definition of the characteristics and attributes of risk-based PIs.

The AEOD staff is making progressive incremental improvements in its computational tools. It does not, however, have a long-range vision of the tools and resources that should be available to support risk-informed and performance-based regulation. We recommend that such a long-range plan be formulated for the development of computational tools.

The AEOD staff plans to enhance the ASP program to provide a more useful experience base for evaluating PRA results. The study of reliability of specific systems is a most important adjunct to these studies. The planned addition to its study list of selected systems that are important to safety is timely. We welcome the opportunity to participate in this important work.

Sincerely,

/s/

T. S. Kress
Chairman

References:

1. Office for Analysis and Evaluation of Operational Data report, "Risk-Based Analysis of Reactor Operating Experience," dated December 15, 1995
2. Memorandum dated March 22, 1996, from C. E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to Office of Nuclear Reactor Regulation Directors and Regional Directors, NRC, Subject: Special Report - Emergency Diesel Generator Power System Reliability 1987-1993, INEL-95-0035 (1 volume)

3. Memorandum dated December 22, 1995, from C. E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to G. Holahan, NRR, D. Crutchfield, M. Hodges and L. Shao, Office of Nuclear Regulatory Research, NRC, Subject: Common Cause Failure Parameter Estimates for Selected Components, INEL-94-0064 (6 volumes)