No. 93-189 December 29, 1993

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received four letter-type reports from its independent Advisory Committee on Reactor Safeguards and a letter commenting on a NRC response to an ACRS recommendation on the reliability of emergency diesel generators. The four ACRS letter reports provide comments on:

1) proposed amendments to NRC's Part 73 to protect against malevolent use of vehicles at nuclear power plants;

2) Thermo-Lag fire barriers;

3) a review of the advanced boiling water reactor final safety evaluation report; and

4) Electric Power Research Institute's advanced light water reactor utility requirements document--Volume III.

In addition, the ACRS has sent two letter reports to the NRC's Executive Director for Operations that comment on the NRC's individual plant examination program and

diversity in the method of measuring reactor pressure vessel water level in the advanced and simplified
boiling water reactor designs.

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Attachments: As stated

December 10, 1993

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

Dear Chairman Selin:

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

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we discussed SECY-93-270, which contains proposed amendments to 10 CFR Part 73 to protect against malevolent use of vehicles at nuclear power plants. Our Subcommittee on Safeguards and Security reviewed this matter during a meeting on November 3, 1993. During this review we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

We do not support the proposal to go ahead with expedited rulemaking, and regret that the issue came to us so late in the process that it is awkward to apply brakes now. But it is never too late.

The stated reason for enhancing the defenses of nuclear power plants against attack through vehicle-borne people or explosives (the staff interprets the word vehicle in the narrow sense, as car or truck) is that the attack on the World Trade Center and the unplanned intrusion at Three Mile Island Unit 1 provide bases for increasing both the Design-Basis Threat and the "Actual Threat." The latter is a euphemism for the best intelligence information available to NRC. We do not believe that either increase is justified by the facts. It is particularly disturbing that the proposed amendments and consequent backfits are on a fast track, lacking the customary analysis, and that the Commission has apparently endorsed this approach.

A threat is always a function of both the level of potential harm to the public and its probability—no matter how challenging it may be to estimate the latter it is always possible. Indeed, an effort to do so can serve to enforce clear thinking. The staff has made no effort to estimate the likelihood of a malevolent intrusion, either before or after the two events cited, but has simply asserted that the risk has increased, and that the increase alone justifies the imposition of new requirements.

One of the more praiseworthy trends in risk management in recent years has been toward effective use of probability as a tool in regulation, whether through risk-based maintenance, cost-benefit analysis of backfits, promulgation of safety goals, or other mechanisms. The Commission has repeatedly endorsed this trend, and it is disheartening to see it so blatantly ignored in this case.

Lest there be a misunderstanding, we do not suggest for a moment that there is no risk, only that there is no basis for the conclusion that it has recently and substantially increased.

The staff also asserts that it is newly concerned about timely warning of the accumulation of explosive materials in sufficient quantity to support an attack. We find that puzzling. All truck bombings for which the explosive material is known to any of us have been conducted with ANFO (ammonium nitrate/fuel oil), and the intelligence agencies have never been able to track the flow. Not only is there prodigious national use of this explosive for blasting (in the order of a megaton per year), but other normal uses of these base materials far exceed this. None of us know of any recent change in the difficulty of tracking this kind of explosive, nor did the staff suggest any.

We believe that the imposition of new requirements to counter the threat of vehicular attack should follow the usual orderly path of analysis for both cost and effectiveness. The latter should include a realistic assessment of the threat, including probabilities. Of course this is hard, but it is not impossible. And it depends on the thoughtful use of the best available intelligence data.

We are unconvinced of the need to rush into rulemaking. If there is special information unknown to any of us, leading to the conclusion that the threat has truly changed substantially for the worse, then there may not be adequate time for rulemaking, and the Commission should invoke its emergency authority. There is little to be said for the present course, too slow if the need is urgent, too fast if it is not.

Additional comments by ACRS Members Peter R. Davis, Carlyle Michelson,

William J. Shack, and Charles J. Wylie are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

Additional Comments by ACRS Members Peter R. Davis, Carlyle Michelson, William J. Shack, and Charles J. Wylie

We support the staff recommendation stated in SECY-93-270. It is our view that the proposed rule represents a prudent and effective step toward enhancing public health and safety.

Reference:
SECY-93-270, Proposed Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants, dated September 29, 1993

December 16, 1993

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

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, in response to the referenced Staff Requirements Memorandum, we discussed with representatives of the NRC staff, NUMARC, and industry the technical differences between NUMARC and the NRC staff on the NUMARC test program related to Thermo-Lag fire barriers. Our Subcommittee on Auxiliary and Secondary Systems discussed this matter during a meeting on November 19, 1993. We also had the benefit of the documents referenced.

At the beginning of our review of the Thermo-Lag fire barrier issue, there were several differences between the staff and NUMARC on how the tests should be instrumented and configured to demonstrate compliance with Appendix R. The differences were in the placement of the thermocouples, whether or not cables should be used in the cable trays during testing, and in post-test evaluation of the cable condition. NUMARC has now agreed to use the thermocouple placement suggested by the staff, and the staff appears to have agreed to some testing with cables in the cable tray. How the test results will be used remains open.

The principal concern of the staff is that the limited number of tests will not yield enough data for extrapolation to the large number of specific configurations needing evaluation. The difficulty is compounded by incomplete characterization of the thermophysical properties of Thermo-Lag. The data from the planned tests can be made much more broadly applicable by additional temperature measurements and engineering analysis. In particular, we recommend that the Thermo-Lag cold side surface temperature be measured and that several identical Thermo-Lag configurations be tested with different cable loadings, including no cable. The resulting data and analysis should allow plant-specific cabling and ampacity factors to be dealt with. It should also be possible to resolve NUMARC concerns about excessive conservatism.

Thermo-Lag provides protection from a fire, in part, by material ablation. This suggests to us that aged material may not perform as well as new material. We recommend that at least one test be duplicated with in-service aged Thermo-Lag.

Our interest in fire protection goes beyond the Thermo-Lag issue. We are concerned about the use of standards and practices that are based on fire

protection standards developed for other industries. Their utilization for nuclear power plant application should be specifically evaluated. The move towards risk-based regulation leads us to question present fire risk methodologies, and the adequacy of fire science talent within the agency. We look forward to being kept informed by the staff and NUMARC when they reconsider current fire protection regulations.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

References:

1. Staff Requirements Memorandum, dated November 15, 1993, to J. M. Taylor, EDO, and J. T. Larkins, ACRS, from S. J. Chilk, Secretary, regarding the October 29, 1993 Commission Briefing on Thermo-Lag
2. Memorandum, dated November 10, 1993, to J. T. Larkins, ACRS, from A. Thadani, NRR, regarding ACRS Subcommittee Meeting on Thermo-Lag
3. Memorandum, dated October 8, 1993, for the Commissioners from J. M. Taylor, EDO, Subject: Quarterly Updates of the Thermo-Lag and Fire Protection Task Action Plans

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December 15, 1993

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

Dear Chairman Selin:

SUBJECT: ACRS REVIEW OF THE ADVANCED BOILING WATER REACTOR FINAL SAFETY EVALUATION REPORT

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed the schedule for completing our review of the NRC staff Final Safety Evaluation Report (FSER) for the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) Standard Safety Analysis Report (SSAR). Previous schedules for our review of the ABWR were discussed in the referenced documents.

Our review of the FSER for the ABWR started with an ABWR Subcommittee meeting in October 1993, followed by another meeting in November. (During earlier Subcommittee meetings going back to 1989, we had reviewed ABWR/SER material.) Additional meetings are planned for December and January as advance copies of final draft material become available. Our Subcommittees on Computers in Nuclear Power Plant Operations, Design Acceptance Criteria, Severe Accidents, and Probabilistic Risk Assessment have met to review FSER areas of special interest to them.

The version of the FSER that we are reviewing is thought to cover most GENE submittals through Amendment 31 of the SSAR. This amendment was a reissuance of the complete SSAR in July 1993. Since then, GENE has issued an extensive revision as Amendment 32 and has just issued Amendment 33 on December 7, 1993. The staff intends to update its FSER through Amendment 33 during January 1994.

It appears likely to us that an additional SSAR amendment (beyond 33) will be needed to take care of a significant number of items that we have brought to the attention of GENE during and since our previous reviews of the SSAR (which were based on various earlier amendments). These items include numerous errors and inconsistencies in the SSAR and the absence of certain key information that we believe will be essential to obtaining a favorable Committee report. Some of these items were accommodated in Amendment 32. Items brought to the attention of GENE by late November might be covered in Amendment 33. Additional items are likely to surface during the December and January Subcommittee meetings. All of our items must be closed with a final amendment issued by mid-February, reviewed expeditiously by the NRC staff, and considered

by our ABWR Subcommittee at a meeting scheduled for March 9, 1994. We intend to complete our review and issue a final report only after the FSER is revised to reflect the final amendment to the SSAR.

On this basis, our ABWR Subcommittee will prepare, for full Committee consideration in March, a draft report on those portions of the ABWR application which concern safety. Barring untimely receipt of needed information or completion of the FSER revision, we expect to issue a final report to you in April 1994.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

References:

1. SECY-93-097, dated April 14, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Integrated Review Schedules for the Evolutionary and Advanced Light Water Reactor Projects
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule
3. ACRS Report dated March 18, 1993, to Chairman Selin from Paul Shewmon, ACRS Chairman, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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December 23, 1993

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

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME III PASSIVE PLANTS

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed the staff Final Safety Evaluation Report (FSER) for Volume III of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Passive Plants. Our Subcommittee on Improved Light Water Reactors held a meeting on October 6, 1993, to review this subject. Our final deliberations on this matter occurred during our 404th meeting, December 9-11, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The overall objective was to establish utility industry policy along with top-tier technical and operational criteria for evolutionary and passive plant designs that would facilitate standardization and combined licensing. The intent of the program was to resolve as many of the policy, technical, and licensing issues as could be identified before specific plant designs were to be submitted, or approved. The remaining specific detailed technical and operational issues were to be resolved during consideration of detailed design information on specific plant design submittals. The program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without

delays. The approach was to use utility experience to establish design philosophy, produce design criteria and guidance to achieve the objective, and to address the policies and regulations of the NRC.

The EPRI ALWR URD is a compendium of technical requirements for the design, construction, and performance of ALWR nuclear power plants for the 1990s and beyond. The URD consists of three volumes:

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant.

- Volume II, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for evolutionary nuclear power plants.

- Volume III, "ALWR Passive Plant," consists of 13 chapters and contains utility design requirements for passive nuclear power plants.

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We discussed development of the EPRI URD program and the NRC staff reviews during numerous Subcommittee and full Committee meetings. We previously presented our comments to the Commission pertaining to the FSER for Volume II by our report of August 18, 1992.

Volume III is similar to Volume II and many chapters are identical except for the features, requirements, and those policy, technical, and licensing issues unique to the passive plants. Although the Standard Review Plan (SRP) was used by the staff as guidance, the level of detail in the URD did not permit a verification of adequacy. (The SRP was written to support the review of the final safety analysis reports on specific plant designs for which a significant amount of design and construction information is normally available.) The staff conducted its review with the understanding that the EPRI design criteria would meet all current regulations, except where deviations were identified. The staff review of the URD focused on determining whether the EPRI criteria conflict with current regulatory requirements.

In addition, the staff identified a number of policy, technical, and licensing issues which needed resolution in order to complete its review of the ALWRs, including the URD. We provided comments on these issues by our referenced letters. The Commission considered the staff positions on twenty-one of the issues identified in SECY-93-087 pertaining to passive plants.

We believe that the staff has conducted a thorough and comprehensive review. We are in general agreement with the FSER pertaining to Volume III and its conclusion that meeting the URD requirements could result in a reactor design that would not conflict with regulatory guidelines, and that would be responsive to various policy statements. Nevertheless, we are disappointed in the limited technical basis provided for several of the requirements relating to severe accidents – in particular hydrogen control, melt spreading and coolability, and fuel coolant interaction (steam explosion). In addition, we believe additional consideration should have been given to general design criteria for containment to withstand severe accident loads.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

References:

1. SECY-93-087, dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
2. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evalua tion Report for Volume II of the Electric Power Research Institute's Advanced Light Wa ter Reactor Requirements Docu ment, including the following enclosures:
   - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
   - Final Safety Evaluation Report for Volume II, "NRC Review of the Elecric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
3. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13 through Revision 4, dated April 1992
4. Draft Commission Paper, undated, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs
5. Staff Requirements Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary, to James M. Taylor, Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
6. Letter dated November 10, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Draft Commission Paper, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs"
7. Letter dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
8. Letter dated August 18, 1992, from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants
9. Letter dated August 17, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
10. Letter dated May 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
11. Letter dated April 26, 1990, from Carlyle Michelson, ACRSChairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements

December 14, 1993

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

Dear Chairman Selin:

On several occasions we have written you about the staff devotion to "trigger values" in its effort to assure emergency diesel generator (EDG) reliability in the context of the rule on Station Blackout. We have said that the concept is statistically flawed, and in our last letter that it is categorically impossible to demonstrate the reliability of EDGs using these methods.

The attached response by the EDO seems to acknowledge the error, but states that the staff intends to make changes only in the Generic Letter, but not in the Regulatory Guide, because everyone knows the procedure is wrong. We find that a curious and unsatisfactory response. The EDO can doubtless outlast us, but that is hardly a proper remedy for mathematical error.

The EDO's response appears to suggest that the desire for mathematical rectitude is an unnecessary decoration in nuclear regulation. We disagree.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

Attachment:
Letter dated October 29, 1993, from James M. Taylor, EDO, to J. Ernest Wilkins, Jr., ACRS Chairman, regarding ACRS concern over "trigger value" approach proposed by Regulatory Guide 1.160

September 22, 1993

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

Dear Chairman Selin:

SUBJECT: PROPOSED GENERIC LETTER REGARDING REMOVAL OF ACCELERATED TESTING AND SPECIAL REPORTING REQUIREMENTS FOR EMERGENCY DIESEL GENERATORS FROM PLANT TECHNICAL SPECIFICATIONS

During the 401st meeting of the Advisory Committee on Reactor Safeguards, September 9-10, 1993, we reviewed the subject generic letter (GL). During this meeting, we had the benefit of discussions with representatives of the NRC staff and NUMARC. We also had the benefit of the documents referenced.

The staff has informed us that this version of the proposed GL reflects consideration of the comments made by the Committee to Review Generic Requirements (CRGR). The proposed GL has been issued for public comment in accordance with our agreement that this could be done prior to our review.

The proposed GL would allow licensees to request removal of the Technical Specification (TS) provisions for accelerated testing and special reporting requirements for the emergency diesel generators (EDGs). When requesting this license amendment, licensees must, however, commit to implement a maintenance program for monitoring and maintaining EDG performance consistent with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and of Regulatory Guide (RG) 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," that was developed by the staff to provide guidance for complying with the provisions of the Maintenance Rule, 10 CFR 50.65.

In our April 26, 1993 (revised June 24, 1993) report on the draft version of RG 1.160, we noted that:

> On many occasions, we have provided comments on the trigger-value approach proposed by the staff to resolve Generic Issue B-56, "Diesel Generator Reliability." The proposed regulatory guide for implementing the Maintenance Rule explicitly endorses the trigger-value procedure for "monitoring emergency diesel generator (EDG) performance against EDG target reliability levels." It is categorically impossible to demonstrate the reliability of EDGs using this method. We remain strongly opposed to its use for this purpose and continue to recommend that the staff's implementation guidance for the Station Blackout Rule, 10 CFR 50.63, be revised to deal with this issue. When this is done, the regulatory guide should be appropriately revised.

The staff's response was to include a footnote in RG 1.160 which states:

> The triggers are intended to indicate when emergency diesel generator performance problems exist such that additional monitoring or corrective action is necessary. It is recognized that it is not practical to demonstrate by statistical analysis that conformance to the trigger values will ensure theattainment of high reliability, with a reasonable degree of confidence, of individual EDG units.

We do not believe that this footnote satisfactorily resolves our concern. Regulatory Guide 1.160 endorses Section 12.2.4 of NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which, in turn, references Appendix D of NUMARC 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at LWRs." Each of these documents clearly implies that use of the "trigger values and monitoring methods" (as described in Appendix D of NUMARC

87-00, Revision 1) provides an acceptable means of monitoring EDG target reliabilities of 0.95 or 0.975 in accordance with the intent of 10 CFR 50.63 for coping with station blackouts. (See, for example, the language of the first paragraph of the introduction to Appendix D of NUMARC 87-00, Revision 1.) It can't be both ways! We strongly recommend that the staff and NUMARC collaborate in resolving this matter by appropriate revision of these documents.

We have had a longstanding concern that the EDGs at many nuclear power plants are being subjected to excessive and unnecessary surveillance testing and other testing as required by TS limiting conditions for operation, and that such testing may actually be degrading the reliability of these machines. Data for the years 1988 to 1991, provided to us by NUMARC, show that some EDGs are subjected to start testing only 12 to 15 times each year, while other EDGs are tested over 100 times each year.

This disparity in testing frequencies results, in part, from the wide variation in relevant TS requirements that were negotiated with licensees over the years. The fact that this situation has existed for so many years reflects badly on both the staff and licensees with respect to their effectiveness in dealing with an acknowledged problem having safety implications. We believe that this proposed GL is an important step in achieving a more rational testing program.

In addition to our recommendation that RG 1.160 and the NUMARC documents on which this proposed GL is based be revised to reflect statistical reality, we believe that the language of the proposed GL needs improvement. The proposed GL quotes a statement from RG 1.160 that triggers and testing of "problem diesels" will be addressed separately by the NRC. In the next paragraph of the GL, licensee commitments required for approval of the removal of accelerated testing and special reporting requirements from the TS are described, including the need for a commitment to RG 1.160. A statement is then made that these actions are intended to close the issues of triggers and testing for "problem diesels." The staff should clarify this apparent contradiction and state clearly that the former prescriptive requirement for accelerated testing has been eliminated by this proposed generic letter.

Sincerely,

J. Ernest Wilkins, Jr., Chairman, ACRS

References:

1. Memorandum dated August 13, 1993, from J. Larkins, ACRS, to B. Grimes, Office of Nuclear Reactor Regulation, Subject: Proposed NRC Generic Letter "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Plant Technical Specifications"
2. Memorandum dated August 12, 1993, from G. Marcus, Office of Nuclear Reactor Regulation, for J. Larkins, ACRS, forwarding proposed NRC Generic Letter Regarding Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Plant Technical Specifications
3. U.S. NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," June 1993
4. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 1993
5. Appendix D. "EDG Reliability Program" to NUMARC Report, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at LWRs," NUMARC 87-00, Revision 1, August 1991
6. SECY-93-044 dated February 22, 1993, for the Commission from James M. Taylor, NRC Executive Director for Operations, Subject: Resolution of Generic Safety Issue B-56, "Diesel Generator Reliability"
7. Letter dated April 26, 1993 (Revised June 24, 1993), from Paul Shewmon, ACRS Chairman, to Brian K. Grimes, Office of uclear Reactor Regulation, Subject: Implementation Guidance for the Maintenance Rule

8. Letter dated August 7, 1992, from Alex Marion, NUMARC, to Paul Boehnert, ACRS, providing Industrywide Emergency Diesel Generator Reliability Data
9. Letter dated December 18, 1992, from Raymond Fraley, ACRS, to Alex Marion, NUMARC, Subject: Industrywide Emergency Diesel Generator Reliability Performance Data
10. Letter dated March 1, 1993, from Alex Marion, NUMARC, to Raymond Fraley, ACRS, responding to questions regarding Industrywide Emergency Diesel Generator Reliability Performance Data

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December 16, 1993

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

Dear Mr. Taylor:

SUBJECT: INDIVIDUAL PLANT EXAMINATION PROGRAM

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed the status of the Individual Plant Examination (IPE) program and some aspects of the resolution of Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs) by the IPE and the Individual Plant Examination of External Events (IPEEE) programs. These matters were also discussed during a meeting of our Subcommittee on Individual Plant Examinations on November 18, 1993. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We share the conclusion expressed by the staff that the IPE program appears to have exceeded expectations. We are particularly encouraged that (1) licensees are actually using their IPE results to effect safety improvements at their plants and (2) most, if not all, licensees plan to maintain their IPEs as an ongoing current assessment of plant risks (although not required), and to use the results as an important adjunct in making decisions with potential safety implications. We also note that the definition of the database structure to be used in the collection and correlation of IPE/IPEEE program results has received careful attention early in the program. Both the format for entering results into the database and flexible retrieval capabilities will be provided for users. A users handbook is scheduled to be available in mid-1994. We do have two concerns about the IPE process that we will provide later in this letter.

With respect to the IPEEE process, we are in general agreement with the staff approach. We would recommend, however, that the staff consider the possibility of developing some method for converting the qualitative approaches for evaluating external events (such as the Fire Induced Vulnerability Evaluation and the Seismic Margin Assessment methodologies) into quantitative equivalents. This would facilitate determinations of relative significance, provide a more definitive framework for decisionmaking, and aid in an overall assessment of the status of the population of plants with respect to the safety goals.

Regarding the Accident Management program, we expect to comment when the program is more nearly complete. We are favorably impressed, however, with the extensive and constructive interaction between the staff and industry during the development of accident management strategies. We encourage a continuation of this interaction.

Returning to the IPE process, we have two concerns, as follows:

1. In our limited review of several IPE results, we were perplexed by the wide variation in reported values for conditional containment failure probability (CCFP). The values ranged from a few percent to 80 percent,

and this was based on only a few IPE samples. Since in many cases neither public risks nor large release probabilities are provided (nor were they required) in the IPEs, it is difficult to decide how to determine the existence of a containment performance vulnerability. Part of the large CCFP range appears to be related to a variation in the definition of "containment failure." For example, it appears that different views have been taken on whether basemat meltthrough, deliberate venting, or interfacing system LOCA events constitute containment failure. Furthermore, the mode and timing of containment failure, as well as the precursor events postulated to occur during a core melt sequence, can cause vast variations in the estimated atmospheric source term (not computed in many of the IPEs). For example, an early overpressure failure in a PWR is obviously of much greater concern in terms of adverse public impact than a late containment basemat meltthrough. Yet, both failure modes may show up in the IPEs as equivalent contributors to CCFP.

As a result of the inconsistent definition of containment failure and the potential for an exceedingly wide variation in the source term resulting from different containment failures, it is not clear how the staff can draw any meaningful conclusions regarding containment vulnerabilities for an individual plant. We recommend that the staff give this matter additional consideration and try to establish a framework for evaluating containment performance for severe accidents from the IPE information. Part of this might include the formulation of an impact index to describe the relative risk significance of various modes of containment failure described in the IPEs based on equivalent containment failure parameters from the NUREG-1150 results as well as the results from other Level III PRAs.

2. We are concerned that the resolution of safety issues (USIs/ GSIs) by the IPE process is not being tracked and evaluated by the staff. We agree, as we have stated in the past, that the IPE/IPEEE process is an appropriate mechanism for the resolution of those USIs/GSIs that appear to be highly plant- specific. However, we were informed that the IPE reviews are not to be focused on the treatment and corresponding results for the USIs/GSIs which are expected to be included in the IPEs. (This position is consistent with the information contained in your letter to the ACRS Chairman dated September 22, 1993.) We are concerned that inadequate or incorrect models and assumptions might be used in the treatment of the USIs/GSIs. This can obscure the significance of the USIs/GSIs for a particular plant and would not be discovered because of the incomplete review process. We note in this regard that the latest IPE Review Guidance Document does not specify review requirements for USIs/GSIs. The staff previously determined that the USIs/GSIs had potential safety significance (or they would not be safety issues), based on the results of staff research and assessment. It therefore seems important that the IPEs be reviewed specifically to ensure that (a) the USIs/GSIs have been considered by the IPE assessments, and (b) the models andassumptions associated with the treatment of the USIs/GSIs in the IPE be consistent with the staff's understanding of these issues based on the evaluations performed to establish the significance of the USIs/GSIs. It seems to us that this review could be facilitated by incorporating specific USIs/GSIs into the database being created by Brookhaven National Laboratory from the IPE results.

We look forward to continued interaction with the staff on these important topics.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

References:

1. U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, November 23, 1988, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)"
2. U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement 4, June 28, 1991, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)"
3. Letter dated September 22, 1993, from Mr. James M. Taylor, Executive Director for Operations, for Dr. J. Ernest Wilkins, Jr., ACRS Chairman, Subject: ACRS Letter Dated August 11, 1993: Proposed Resolution of Generic Issue 143, "Availability of Chilled Water System and Room Cooling"
4. Memorandum dated November 18, 1993, from Warren Minners, Office of Nuclear Regulatory Research, for John Larkins, ACRS, Subject: IPE Review Guidance Document (Draft Predecisional)

December 16, 1993

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

Dear Mr. Taylor:

SUBJECT: DIVERSITY IN THE METHOD OF MEASURING REACTOR PRESSURE VESSEL WATER LEVEL IN THE ADVANCED AND SIMPLIFIED BOILING WATER REACTOR DESIGNS

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed a proposal, advanced by representatives of the NRC staff, that General Electric Nuclear Energy (GENE) be required to install reactor pressure vessel (RPV) water level instrumentation that is diverse in operation from that presently employed on the Advanced Boiling Water Reactor (ABWR) and Simplified Boiling Water Reactor (SBWR) designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and GENE. We also had the benefit of the referenced documents.

We heard opposing views from the staff and GENE on the need for diversity in the method of measuring RPV water level in the ABWR and SBWR. The staff argues that ". . . two independent and diverse methods for measuring the RPV level should be required because of the importance of RPV level instrumentation to BWRs and because operating experience has shown the potential for failure of redundant level instruments due to common cause." The argument given by GENE is that the ABWR water level instrumentation is rugged, simple, and highly redundant with no known remaining operational problems. GENE further argues that alternate vessel level measurement technologies are unqualified for this application.
The staff has concluded that the differential pressure level measurement system employed in current BWRs provides adequate indication of reactor vessel water level. The staff has also concluded that the presently proposed ABWR level instrumentation meets the minimum requirements of all applicable General Design Criteria. It is the staff's interpretation, however, that this proposed instrumentation may not be in compliance with the relevant post-TMI requirement as codified in 10 CFR 50.34(f).

We do not believe that a case has been made by the staff for a water level indication system in advanced BWRs that is different from that currently used in operating BWRs.

Additional comments by ACRS Members Ivan Catton and Thomas S. Kress are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman, ACRS

Additional Comments by ACRS Members Ivan Catton and Thomas S. Kress

We agree that the present method of measuring vessel water level is sufficient for adequate protection for BWRs and that it is not appropriate to backfit new diversity into existing plants. Nevertheless, an objective of advanced and passive plants is to provide a higher level of safety assurance. We believe that the availability of at least three alternative level measuring methods affords an opportunity to provide this higher level of assurance in this important area. We agree with the staff's recommendation that installation of diverse vessel level instrumentation be required for the ABWR and SBWR designs.

References:

1. Proposed Draft SECY Paper (undated), from James M. Taylor, EDO, for the Commissioners, Subject, Diversity in the Method of Measuring Reactor Pressure Vessel Level in Advanced Boiling Water Reactor and Simplified Boiling Water Reactor (Draft Predecisional)
2. Memorandum dated December 10, 1993, from P. Boehnert, ACRS, for ACRS Members, Subject: ACRS Review of Proposed Requirement for Diverse Vessel Water Level Instrumentation for ABWR/SBWR - Additional Information on Diverse Level Instrumentation for German and Swedish BWR Plants