March 14, 1994
EN 94-023

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OFFICE OF ENFORCEMENT
NOTIFICATION OF SIGNIFICANT ENFORCEMENT ACTION

Licensee: Morgan County Memorial Hospital (EA 93-250)
Martinsville, Indiana
Docket No. 030-12775
License No. 13-17449-01

Subject: PROPOSED IMPOSITION OF CIVIL PENALTIES - $9,750

This is to inform the Commission that a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $9,750 will be issued on or about March 14, 1994 to Morgan County Memorial Hospital of Martinsville, Indiana. This action is based on: 1) the deliberate actions of a technologist in failing to make daily and weekly radiation surveys as required and falsifying NRC-required records to make it appear that the surveys had, in fact, been performed, categorized at Severity Level II; and 2) careless disregard on the part of the same technologist regarding storing and consuming beverages in areas where radioactive materials were used, categorized at Severity Level III. The base civil penalties were escalated 50% because NRC identified the violations. The Commission was advised of this proposal in the enclosure to SECY 94-037, which provides additional details on the violations and proposed enforcement action. Separate enforcement action is being taken against the technologist as an individual (IA 94-002).

It should be noted that the licensee has not been specifically informed of the enforcement action. The schedule of issuance and notification is:

Mailing of Notice	March 14, 1994
Telephone Notification of Licensee	March 14, 1994

The State of Indiana will be notified.

The licensee has thirty days from the date of the Notice in which to respond. Following NRC evaluation of the response, the civil penalty may be remitted, mitigated, or imposed by Order.

Contact: J. DelMedico, OE, 504-2739 J. Lieberman, OE, 504-2741

Distribution:

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

February 23, 1994

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

Dear Mr. President:

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

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

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

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

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

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

ACRS-1558

February 23, 1994

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

Dear Mr. Speaker:

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

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Chairman

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

February 19, 1993

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

Dear Chairman Selin:

SUBJECT: ISSUES PERTAINING TO THE ADVANCED REACTOR (PRISM, MHTGR, AND PIUS) AND CANDU 3 DESIGNS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 393rd and 394th meetings of the Advisory Committee on Reactor Safeguards, January 7-8 and February 11-13, 1993, we reviewed a draft Commission paper on the cited subject. Our Subcommittee on Advanced Reactor Designs also met on January 6, 1993, to discuss this matter. We had the benefit of discussions with representatives of the NRC staff, the Department of Energy, and the preapplicants: Atomic Energy of Canada, Limited, Technologies (AECLT), General Electric Nuclear Energy (GE), and General Atomics (GA). We also had the benefit of the referenced documents.

The draft Commission paper lists ten issues that need policy direction from the Commission for proposed deviations from existing regulations. These deviations arise either because existing regulations are generally specific to light water reactors (LWRs), or because the criteria proposed by the designers of the four reactor types listed are significantly different from those in the existing regulations. The draft paper also classified these ten issues into two categories: (1) those issues for which the staff agrees that departures from current regulations should be considered and (2) those issues for which the staff does not believe a departure from current regulations is warranted at this time. Not all of these issues are relevant to each reactor type; the draft paper contains a matrix identifying plant applicability. The paper contains some general comments and recommendations, as well as specific comments and recommendations on each of the ten issues.

Everything we say is predicated on our understanding of the applicable safety policies, which we would describe as follows:

- The safety objective for the nuclear enterprise was described in the 1986 Policy Statement on Safety Goals, and has not been rescinded. There is no distinction drawn in there between existing plants and new plants.

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- The ACRS has recommended that the principal use of the goals be to judge the effectiveness of the entire enterprise, including regulation, in producing a plant population consistent with the goals. The Commission has never rejected that view.

- If the industry chooses to do better, we can only applaud its zeal, but ought not to stifle initiative by transforming initiatives into requirements.

Our views on the various items in the referenced draft paper are given below.

GENERAL COMMENTS

1. We find that the identified issues are important and that the staff should receive guidance from the Commission. (There are other policy issues affecting these reactor designs that are being addressed in connection with the evolutionary and passive LWR designs.) There may well be additional policy issues that appear during the preapplication review process. The staff has committed to identify any such issues in subsequent Commission papers.

2. The staff has grouped these ten issues into the two categories described above. We note that all of the affected preapplicants who appeared before us would treat Issue I (Control Room and Remote Shutdown Area Design) as a Category 1 issue, whereas the staff proposes it as a Category 2 issue. We will discuss this difference of opinion below in our opinion on Issue I.

3. For Category 1 issues, the staff proposes more conservative alternatives than the preapplicants propose, in order to account for uncertainties associated with the conceptual design. We are concerned that such an approach might well freeze an unnecessarily large degree of conservatism into the designs, and the preapplicants would have great difficulty persuading the staff to relax this conservatism on the basis of more precise information available in the final design.

4. We support the staff recommendation that "a prototype CANDU 3 is not required for design certification."

5. We support the staff intention to notify the Commission if its position on any of these ten issues should change, or if new issues are identified.

6. We have no objection to the staff recommendation that the highest priority be given to issues that are applicable to the PRISM design.

7. We understand and sympathize with the staff recommendation to defer decisions on generic rulemaking on these ten issues. Nevertheless, we urge the Commission to address these decisions in the near future. (The generic rulemaking question may arise in connection with passive LWR designs.)

8. In several places in the draft Commission paper, there occurs qualitative language, e.g., "appropriate conservatisms" or "credible severe accidents." This language must ultimately be translated into quantitative guidance. We believe that the quantitative guidance is, to a large measure, policymaking, and should not be relegated to low-level reviewers.

<u>SPECIFIC COMMENTS</u>

<u>Category 1 Issues</u>

A. Accident Evaluation

The staff proposal to develop a single approach with certain specified characteristics appears reasonable. We would like to review that approach when it is ready. We believe, however, that the staff should identify at an early stage quantitative guidelines and criteria for accident selection and evaluation. We note that AECLT has taken exception to some of the statements in the draft Commission paper that relate to its approach to this issue. We believe that this disagreement can be resolved by AECLT and the staff.

B. Source Term

The staff proposal to base the source terms on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded. We note that the staff is now developing for LWRs a revision to the TID-14844 source term. It will be appropriate for the staff to consider using the newer approach when it develops source terms, and to take specific account of the unique features of each of the reactor types.

C. Containment

The staff proposal "to postulate a core damage accident as a containment challenge ..." appears reasonable. We would like to review the list of postulated accidents when it is ready.

D. Emergency Planning

The staff proposes that advanced reactor licensees be required to develop offsite emergency plans which will include a requirement for onsite and offsite exercises. This proposal appears reasonable under the present circumstances, except that we would follow existing LWR guidance that permits the omission of offsite exercises when it can be shown that the design would preclude any accidental release exceeding the EPA Protective Action Guides. The staff has agreed to consider, after a review of Accident Evaluation (Issue A, above), whether some relaxation from current requirements may be appropriate. We urge that work on Issue D be closely correlated with work on Issues A and B, in order to avoid unnecessary conservatism.

E. Reactivity Control System

The staff proposal that the absence of control rods need not disqualify a reactor design, provided that an applicant can show a level of safety in reactor control equivalent to that of a traditional rodded system, appears reasonable. We note that this issue is applicable only to the PIUS concept, and that we have not yet had the benefit of presentations by the PIUS designers.

F. Operator Staffing and Function

The staff intends to review the justification for a smaller crew size by evaluating the function and task analyses for normal operation and accident management. This intention appears reasonable, although we believe that particular attention needs to be given to multiple module designs. We note that this issue is related to a similar issue for passive reactors. We believe that the Commission policy should be the same for the advanced reactors and CANDU 3 as it is for the passive reactors.

G. Residual Heat Removal

The staff belief that reliance on a single, completely passive, safety-related residual heat removal (RHR) system may be acceptable appears reasonable, although we would have liked to see the criteria to be used by the staff in deciding acceptability. We agree with the staff that NRC regulatory treatment of non-safety-related backup RHR systems for these reactors should be consistent with design requirements (not yet identified) for passive LWRs.

H. Positive Void Reactivity Coefficient

We agree with the staff that the existence of a positive void reactivity coefficient is a significant concern, but that it should not necessarily disqualify a reactor design. The burden of showing that the consequences of those accidents that would be aggravated by a positive void reactivity coefficient are either acceptable or could be satisfactorily mitigated by other design features surely falls on the preapplicant. On the other hand, the staff should state the criteria it will use to judge "acceptable" or "satisfactorily."

Category 2 Issues

I. Control Room and Remote Shutdown Area Design

We do not agree with the staff decision to treat this issue as a Category 2 issue, and the concomitant recommendation to apply current LWR regulations and guidance until passive LWR policy in this area is finalized. We believe that this issue should be a Category 1 issue, and that the preapplicants should accept the burden of convincing the staff that a proposed design is satisfactory, according to some criteria that should be specified by the staff.

J. Safety Classification of Structures, Systems, and Components

This issue is relevant only to the MHTGR concept. GA makes a persuasive case that the MHTGR is sufficiently different that the LWR criteria for identification of safety-related structures, systems, and components should not arbitrarily be applied to the MHTGR. We concur with this view and believe that Issue J should also be classified as a Category 1 issue. This would not preclude coordination of the policy for passive reactors with the policy for the MHTGR.

Our interest in all these matters continues. We would like an opportunity to review any significant change in staff or preapplicants position, as well as any significant developments in the implementation of the policies.

Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding issues related to the MHTGR.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Memorandum dated December 2, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, transmitting Advance Information Copy of Forthcoming Commission Paper - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements
2. Letter dated January 28, 1993, from David P. Hoffman, Gas-Cooled Reactor Associates, Management Committee, for D. M. Crutchfield, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues Concerning the Preapplication Reviews of Advanced Reactors
3. Letter dated January 25, 1993, from Peter M. Williams, Department of Energy, to J. Donohew, Office of Nuclear Reactor Regulation, NRC, commenting on the draft Commission Paper
4. Letter dated January 25, 1993, from N. Grossman, Department of Energy, to S. Sands, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues and Schedules Concerning the Preapplication Reviews of Advanced Reactor and CANDU 3 Designs

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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's progress in defining the regulatory requirements for digital instrumentation and control systems. During this meeting, we had the benefit of discussions with members of the NRC staff.

We have now had a long series of meetings, and have heard from many relevant people, but by no means all. To some extent our input has been biased in the direction of people, groups, and organizations who have experienced problems, and we have not heard from the legions of organizations who have successfully made the move into the computer world. It is important not to develop a tabloid mentality about new technology, i.e., aberrations from the norm treated as if they were the norm.

A first observation is that many of the anecdotes about catastrophic failures of major computer systems refer to systems far larger than those of interest here. Even the software systems on the C-17 aircraft, written in nearly a dozen languages for nearly a dozen machines, are far larger than any of relevance to the nuclear business. The Strategic Defense Initiative dispute is even less relevant. So we have to maintain perspective about scale.

A second observation is that computerization provides an opportunity, not a threat. The extraordinary reliability of electronic systems (unless abused), their potential for continuous and extensive self-testing in real time, their potential for relatively painless upgrades as experience accumulates, their ability to cover an enormous function space and to accommodate unseemly amounts of input data, their remarkable immunity to wear

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(few, if any, moving parts)—all these provide the potential for safety enhancement. Much of our input from the staff has been devoted to the negative aspects of computerization, as if it were a disease to be kept in check.

A related observation is that the transition to computerized operation, control, instrumentation, support, recordkeeping, and maintenance procedures and records, is inevitable. The job of the NRC is not to manage or resist the transition, but to maintain a reasonable level of assurance that it is accomplished with proper accounting for the impact on safety. With any reasonable use of the technology the impact is expected to be large and positive.

The regulatory issues we have isolated in our series of subcommittee meetings fall broadly into two categories. One is a consequence of lack of nuclear regulatory experience with modern electronics, especially computers, leading to both extraordinary conservatism relative to unfamiliar accident sequences, and the application to a new technology of review methods and nomenclature derived from old habit and experience. The second is a collection of genuinely new problems associated both with the complexity of the new technology and with the consequent difficulty of assessing (as distinguished from assuring) its level of safety. We deal with these in order.

Failures of computerized systems (excluding fans, hard disks, and other mechanical components) do not follow the traditional bathtub curve of infant mortality, stable performance, and then wearout. Electrons don't wear out. Both in electronic hardware and software there tends to be a period of infant and young adult mortality (to which we will return), with performance and reliability gradually improving with time simply through natural selection—bugs are ironed out through experience and through extensive testing. There is no later period of wear, so there is no place for the regulatory and maintenance procedures associated with that part of the reliability pattern. Further, self-testing can provide constant assurance of full functionality of the electronics.

As a consequence, however, there has been little progress in applying the methods of probabilistic risk analysis, on which we have become so heavily dependent for mechanical, hydraulic, and electromechanical systems, to computer systems. Indeed the semiconductor components of the computerized systems are inherently so reliable that hign-temperature life-testing is the only means available, in most cases, for generating any failures at all. Whereas one can generate probabilities for the existence of perinatal defects, there is no such thing as a probability per unit time for the development of disease. Nor does in-service inspection play the same role.

These are important points, because the concepts of reliability and reproducibility differ, and the testing and verification procedures used depend on which is to be assured. A mechanical component with a presumed reliability of 10^{-3} failures per demand can be tested a few thousand times to assure that level of reliability, but a software-based system with a hidden bug that will be revealed in the event of an unlikely input configuration can be tested without failure until the cows come home, but will still always fail with that particular input. Interest has therefore to be directed at the probability that there is such a hidden bug, and the probability that some other circumstance may generate the unfortunate input. Neither of these probabilities will be discovered by repetitive testing under normal conditions. Randomized input testing can tell one something about the former probability, but not the latter. It is therefore misleading to bandy failure probabilities around, as if they had the same meaning as they do for familiar mechanical and electrical components. It also makes the direct comparison of computerized system reliability with the reliability of older technology more difficult.

These and other considerations mandate a format adjustment for the regulatory system, and such changes tend to be painful. What we have seen here is an unfortunate effort to cling to the old ways, to the point of asking that all digital systems have analog backups—not because the latter are better or more reliable, but because they are more familiar to the regulator and therefore easier to regulate. That alone could place an unwarranted burden on those seeking to improve safety by updating technology.

The second category of issues follows from the undoubted fact that computerized systems do indeed introduce unfamiliar failure modes, which require both recognition and palliative measures. Too much attention appears to have been concentrated on a microcosm of the more recognizable of these matters, specifically vulnerability of digital systems to electromagnetic interference (a subject on which there is enormous military expertise, largely untapped by the NRC staff), and the fact that replicated defective software (like replicated defective hardware) can be the source of common-mode failures. Both of these are real issues, but, in our judgment, not the central ones.

Let us first consider software issues. The literature is full of examples of cases in which carefully written and tested software still contains errors. Indeed it is doubtless true, though in principle unprovable, that any large program that has not undergone a formal verification and validation (V&V) contains yet undiscovered errors. Lest there be confusion, it is well to be quantitative about the problem of implementing a function in software.

The simplest of all digital programs might generate a logic function, a mapping that accepts a number of binary inputs (say n) and generates a single binary output—a signal that might, in turn, activate a pump or a valve or some other sequence of events. Such a logic function has 2^n possible input states, over a thousand for n=10 and over a million for n=20. These are not unreasonable numbers of input states, because the input of a single number to one percent accuracy requires seven (usually more) binary inputs. Since each input state can have either output state (on/off), that means that even a modest eight-input binary converter of this sort can represent 2^{256} or 10^{77} different logic functions. A defect (either hardware or software) can change the desired function into any of the others. It is therefore reasonable to expect to test the system to make sure that it performs as designed, but not reasonable to expect to explore, by brute force, all consequences of all possible defects. The point is only strengthened if one has more complex outputs than just a single bit.

If, therefore, the requirements specified for the system describe the full mapping of the input space to the output space, special methods will be required to verify that this has been accomplished correctly. Such methods exist, and are applicable to relatively simple software packages. When formal V&V is possible, it provides assurance that the code, as written, correctly implements the formal specifications laid upon the design. When it is not possible (because the code is too long or too complex), there are many alternatives, but none of them provides the kind of assurance of code fidelity that is provided by formal V&V.

There appears to be a consensus among the experts we have consulted that the safety-related software in nuclear power plants is within reach of formal V&V methods, and that the potential for serious error lies more in incorrect expression of the specifications than in incorrect programming. Formal V&V can assure that the code correctly expresses the specifications, but not that the specifications are correct. In either case, it would appear that the staff emphasis on the possibility of common-mode errors in code segments used in different parts of the instrumentation and control system is misdirected. We continue to see an urgent need for staff augmentation with people experienced in thinking in the terms outlined above.

We believe that the experience of other industries that have accepted the progress has been characterized, almost without exception, by increases in efficiency and reliability, and by concomitant decreases in cost. (While the latter is not the NRC's business, it remains true that resources and attention released from unproductive safety concerns may, at least in part, find their way to better use.) There are genuine safety issues in this transition, of which one unfamiliar one is surely the requirement, in order to generate verifiable software, for precise no-nonsense attention to the specification of the functions to be implemented by the software.

The gist of our concerns is that the regulatory procedures developed during the decades preceding the full flowering of the electronic revolution (which may not yet have occurred) are inappropriate to the regulation of computerized functions in nuclear power plants. (This is true for both hardware and software—too much emphasis on the distinction is not helpful.) As a consequence, the staff has been dealing with the problems that have shown up so far on an ad hoc basis, applying methods created for each problem, with little underlying methodology. That has resulted in such distractions as the analog-to-digital conversion problem, the overemphasis on electromagnetic interference problems, the singling out of software common-mode failure as a central issue, etc., all without a framework into which the broad issues of regulatory emphasis and consistency can be fitted. We can cavil about the specific staff approaches to each of these, but the central issue is that neither the staff nor the Commission has established what could be described as a standard review plan or even a regulatory guide that could help both the staff and the industry know what is expected of them. A statement of the applicable standards ought to precede, not follow, their application. Without such a definition of objectives, coherence is an inevitable victim.

What, then, do we recommend? We frankly doubt that a coherent and effective review plan for computerized applications in nuclear power plants will be produced by the staff, the Commission (whose job is at a higher policy level), or the Committee (which is limited in both resources and expertise). Still, if one believes (as we do) that it needs to be done, it will be necessary to bring in outside help. It was in that context that we initiated our long series of subcommittee meetings on the subject. Our recommendation is that a workshop and study (with a charter to produce such a plan) be commissioned to be done by the National Academies of Sciences and Engineering. To derive maximum benefit from such a study, there should be appropriate participation by key senior members of the staff.

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

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

We agree with most of the technical observations made in this report. However, we disagree with the report's recommendation that a workshop and study be undertaken by the National Academies of Sciences and Engineering for the purpose of developing a review plan for computerized applications in nuclear power plants. Contrary to the view of our colleagues, we believe that the staff and its consultants are making satisfactory progress toward developing a "coherent and effective" review plan. Ideally, such a plan should have been developed in advance of the receipt of applications for the use of this rapidly changing technology. As a practical matter, it has been necessary for the staff to interact with the first group of applicants proposing computerized systems in order to gain an understanding of these systems. This has been a necessary first step before a generic review plan can be developed. Our view is that the proposed National Academies of Sciences and Engineering workshop and study would add little to the process of developing a staff review plan at this point in time.

We note that the staff has attended the series of ACRS subcommittee meetings on computerized applications in nuclear power plants that form the basis for this Committee report. In addition, the staff is planning to sponsor a workshop this fall and plans to obtain ACRS feedback on speakers and topics to be covered.

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

March 19, 1993

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

Dear Chairman Selin:

SUBJECT: HUMAN PERFORMANCE IN OPERATING EVENTS

During the 391st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1992, we discussed with representatives of the Office for Analysis and Evaluation of Operational Data (AEOD) a draft of the AEOD study entitled, "Operating Experience Feedback Report - Human Performance in Operating Events." (This study was issued as NUREG-1275 in December 1992.) Representatives of NUMARC provided comments on the draft of this study during our meeting. We also discussed this matter during our 395th meeting, March 11-12, 1993. We had the benefit of the documents referenced.

This study was conducted over a 2 1/2-year period and involved 16 onsite visits by multidisciplinary teams led by an AEOD staff member for the purpose of evaluating human performance during selected nuclear power plant events. The study focused on factors that influenced operator performance during a wide variety of plant events. AEOD estimates that these events represent approximately 30 percent of the events that challenged operating crews during this 2 1/2-year period. The study summarizes each event and the findings that the teams made, provides observations discerned from related events, and presents conclusions concerning overall human performance. These conclusions fall into four categories of human performance issues: control room organization, procedures, human-machine interface, and industry initiatives. Finally, the study attempts to compare the "latent factors" among these 16 events.

Five of the 16 events studied were also the subject of Augmented Inspection Teams (AITs). We believe that a number of the remaining 11 events were of sufficient significance from a human and organizational performance point of view to have warranted an AIT effort. During our meeting with the AEOD staff we commented that the final version of the study should address this issue, since it may be a weakness in the approach being used by the Office of Nuclear Reactor Regulation (NRR) and the Regional Offices in systematically analyzing and evaluating human performance in operating events. AEOD did not explicitly deal with this issue in the final version of the study.

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We have been critical of AEOD in the past for its reluctance to discuss the performance of NRC staff organizations in the course of carrying out studies of this nature. It continues to be our view, as discussed under Summary and Conclusions below, that this should be a necessary part of AEOD studies of this nature.

The Analysis section of the study (Section 3.0) contains a number of observations and conclusions that we believe are of importance from a nuclear safety perspective. We have the following comments on this section of the study:

- Control room organizational weaknesses were observed in the response of some operating crews to emergency situations (Section 3.2). This matter should receive prompt attention by the staff, with appropriate involvement of NUMARC and/or the Institute of Nuclear Power Operations. The requisite organizational factors approaches needed to deal with emergency situations should be well understood at this stage of maturity of nuclear power plant operations. In addition to the lessons learned in response to actual emergency situations, the staff and licensees have had numerous opportunities to observe and learn from operating crew response during requalification examinations and emergency plan exercises. We recommend that the weaknesses observed be corrected on an expedited basis.

- We are concerned by the two events in which engineered safety features (ESFs) were bypassed (Section 3.3.4). (Neither of these events was raised to the level warranting an AIT and, in one of these cases, the ESF was bypassed without the knowledge of the shift supervisor.) It is not clear from the study if these events were investigated appropriately by the Regional Offices. We believe that occurrences of this kind may represent a serious "safety culture" problem within the licensee organization. The staff should thoroughly review licensee corrective actions for events of this nature to ensure that the <u>real root causes</u> of the events have been dealt with in a manner that will prevent their recurrence. We do not believe that it is sufficient for the licensee to state in its licensee event report (LER) that the control room operator was reprimanded and provided with remedial training; the licensee needs to thoroughly evaluate and correct any "safety culture" issues raised by such events. However, we caution against the staff assuming the role of "de facto management" by prescribing, as opposed to reviewing, licensee management actions.

- We are concerned by the statement in Section 3.5.1 that licensees had prepared an LER "in almost every case" but that "In some cases, it was difficult to tell that the reports (LERs) described the same event. It appears in these cases that the licensee failed to consider the human performance aspects of the event or failed to include that information in the report." During our meeting with the staff, we suggested that the draft study would be strengthened by including a discussion of the completeness of each associated LER with the evaluation of the individual events. We also suggested that a more detailed evaluation be made of this apparent weakness in the present LER program. AEOD chose not to follow our suggestions.

Summary and Conclusions

We believe that the AEOD study has been useful in focusing the attention of NRR and the Regional Offices, as well as that of the industry, on human and organizational performance issues. We agree with AEOD's plan to continue this activity (as described in Section 4.0 of the study) until these issues have been effectively addressed. As discussed above, we recommend that the Commission provide policy direction to AEOD on the matter of its charter, with respect to evaluating the performance of NRC staff organizations in the course of carrying out studies of this nature.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1275, Volume 8, "Operating Experience Feedback Report - Human Performance in Operating Events," December 1992
2. SECY-92-407, dated December 9, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: The Independent Role of the Office for Analysis and Evaluation of Operational Data in the Assessment of Operational Experience and the Investigation of Operational Events

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

April 22, 1993

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

Dear Chairman Selin:

SUBJECT: DEFINITION OF A LARGE RELEASE FOR USE WITH SAFETY GOAL POLICY

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the staff's recommendations in regard to the definition of a large release related to the implementation of the Commission's Safety Goal Policy. During this meeting, we had the benefit of discussions with members of the NRC staff and of the document referenced.

In the draft Commission paper and in the presentation to the Committee, the staff expressed its belief that the development of the definition of a large release is no longer practical or useful and, therefore, it is requesting Commission approval to terminate efforts in this area. We believe the staff has made a conscientious effort with this activity and we agree with its basic conclusions. Our views are as follows:

1. A large release definition would either represent a replacement for the existing safety goals or, if made consistent with the quantitative health objectives (QHOs), would be redundant and unnecessary.

2. New guidelines being developed for implementing the Safety Goal Policy within regulatory analysis and issue prioritization processes adequately meet the originally perceived need for a large release component of the safety goals. These utilize a core damage frequency (CDF) and a conditional containment failure probability (CCFP).

3. Plant performance objectives, viz CDF $\leq 10^{-4}$ and CCFP ≤ 0.1, provide an easily understandable and adequate surrogate for the QHOs and provide quantitative prioritization for two basic aspects of defense in depth (prevention and mitigation). These could help ensure that a plant does not end up with great core protection but marginal containment performance.

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We support the recommendation that the Commission approve the staff's proposal to terminate its effort to develop a definition of a large release.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Reference:
Memorandum dated March 11, 1993, from Warren Minners, Director, RES/DSIR, for John T. Larkins, Acting Executive Director, ACRS, Subject: ACRS Review of Draft Commission Paper on Large Release Determination, w/Enclosure

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

April 26, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-087, "POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the NRC staff positions, delineated in SECY-93-087, on policy, technical, and licensing issues pertaining to evolutionary and advanced light-water reactor designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We have discussed these issues during several of our previous meetings and provided comments and recommendations in the reports referenced.

We are in general agreement with the staff's positions in SECY-93-087; however, we have concerns regarding some issues and offer our comments and recommendations as follows. (The section titles and letter designations correspond to those in SECY-93-087.)

I. SECY-90-016 ISSUES

E. Fire Protection

In our April 26, 1990 report, we pointed out that redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommended that the proposed fire protection enhancements include separation of environmental control systems (i.e., separate heating, ventilating, and air conditioning (HVAC) systems for each train). The staff responded by conceding that separate HVAC arrangements may be needed, although other options may be available to the designer. The Commission endorsed the staff's response.

We remain concerned that a common normal ventilation system (such as that proposed for the ABWR) will be difficult to design to prevent the effluent from a postulated accident in one train of engineered safety features from reaching essential mitigating equipment in the other trains and

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creating conditions that exceed their environmental qualifications. Of particular concern is the capability of ventilation dampers to isolate the effects of high energy pipe ruptures in confined compartments served by the common HVAC system.

G. Hydrogen Control

The staff claims that it has sufficient basis for understanding hydrogen behavior to go forward with licensing criteria. It has not been demonstrated to us that this basis is as extensive, or applicable, as the staff believes. Further, the AP600 and ABB-CE System 80+ designs have containments that are more susceptible to significant damage from hydrogen detonation than most existing and evolutionary plants. This requires that the licensing criteria for this issue be reconsidered.

H. Core Debris Coolability

The staff has weakened the position taken in SECY-90-016 by not requiring that the core debris be adequately quenched. We believe that the present criterion for coolability, namely a cavity floor area greater than $0.02m^2/MWt$, is not soundly based. We recommend that the staff validate containment response to core-on-the-floor accident sequences by independent analyses using, for example, MELCOR, or CORCON and CONTAIN.

J. Containment Performance

We agree with the requirement that containment stresses not exceed ASME Code Service Level C for metal containments, but it is not clear how electrical penetrations through the containment should be considered. Such penetrations utilize nonmetallic electrical insulation as a portion of the containment boundary and need further consideration.

L. Equipment Survivability

We agree that passive plant design features provided only for severe accident mitigation need not be subject to the environmental qualification requirements of 10 CFR 50.45. We believe, however, that such mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the timespan for which they are needed.

II. OTHER EVOLUTIONARY AND PASSIVE DESIGN ISSUES

Q. Defense Against Common-Mode Failure in Digital Instrumentation and Control Systems

The staff's second recommendation is that the vendor or applicant analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). We recommend that the scope of this assessment include consideration of common-mode failures during all events postulated in the SAR (e.g., fire, flood, pipe rupture, and extensive loss of essential power sources) and not be restricted to those events discussed in Chapter 15, "Accident Analysis."

T. Control Room Annunciator (Alarm) Reliability

The staff's basic recommendation is that the Commission approve the position that the alarm system for ALWRs meet the applicable EPRI requirements for redundancy, independence, and separation. These requirements do not include the use of Class 1E equipment and circuits. The staff also seeks approval of an additional position that goes beyond the EPRI requirements. This position is that "alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits." We believe that the staff needs to provide c arification and additional justification for this position.

Collectively, our identified issues represent a significant array of incompletely addressed concerns. We urge that they be addressed on a timely basis to ensure their early consideration by the design teams.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. SECY-93-087, dated April 2, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs

2. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, March 18, 1993
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," September 16, 1992
4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
5. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, August 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, May 13, 1992
7. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactors Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990

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

April 27, 1993

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

Dear Chairman Selin:

SUBJECT: REVIEW OF ORGANIZATIONAL FACTORS RESEARCH PROGRAM

During the 392nd, 394th, 395th and 396th meetings of the Advisory Committee on Reactor Safeguards, December 9-11, 1992, February 11-13, March 11-12, and April 15-17, 1993, respectively, we discussed the Office of Nuclear Regulatory Research (RES) budget for the human factors research program and SECY-93-020, "Review of Organizational Factors Research." In addition, during our February 11-13, 1993 meeting, representatives of the NRC staff and two of the contractors involved in the organizational factors research program (Brookhaven National Laboratory and University of California at Los Angeles) discussed their work. (The other contractors are the Pennsylvania State University and the Accident Prevention Group, Inc.) We also had the benefit of the document referenced.

Members of our Human Factors Subcommittee and two subcommittee consultants attended the November 12, 1992, senior staff management workshop on the organizational factors research program.

ACRS has followed this program since it was revived in 1987. SECY-93-020 provides the results of the comprehensive review performed by RES of its organizational factors research program and a description of changes to be made to the program as a result of this review. In the Summary Section of this SECY document, RES concludes that

> there is a relatively low cost-effectiveness in continuing regulatory research beyond FY 1993, until it is determined that organizational factors can be reliably integrated into PRA models. RES is meeting with NRR to coordinate further development of human reliability analysis modeling of organizational factors for PRA. It is possible that this further effort will continue at a low level of funding in FY 1994.

We were told that RES does not, at this time, propose to fund additional organizational factors research beyond FY 1993. We also learned from our discussions with RES representatives that its

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Nuclear Safety Research Review Committee had not reviewed and provided comments on the need for continuing this program prior to the issuance of SECY-93-020.

After extensive deliberations, we have been unable to arrive at a consensus with respect to the continuation of this research activity. We plan to take this matter up again when NRR completes its user needs evaluation with respect to organizational factors research.

Additional comments by ACRS Members James C. Carroll, Ivan Catton, Peter R. Davis, and Robert L. Seale are presented below.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Additional Comments by ACRS Members James C. Carroll, Ivan Catton, Peter R. Davis, and Robert L. Seale

We believe that the present organizational factors research effort should be continued to the point where a set of useful products becomes available for trial use by the staff and the nuclear utilities. Our reasons for this view are summarized below.

The Relationship Between Organizational Performance and Safety - The Historical Perspective Section of SECY-93-020 states that "poor organizational performance can be a major contributor to safety significant events and that there is a need for an improved technical base for determining the impact of organizational performance on safety." We agree and further believe that this is one of the most important safety issues presently facing the nuclear power industry. The industry knows how to design extremely safe plants from a hardware point of view. However, operating experience indicates that there are many outstanding questions with respect to the ability of the nuclear utilities in the U.S. (and worldwide) to safely manage the operation and maintenance of both operating and future nuclear power plants. The organizational performance of the NRC staff is also of concern to us in that it can have an impact on the safety of the regulated industry.

We note that the SECY paper describes the organizational factors research programs being carried out by the regulatory authorities in Sweden, the UK, and France. This raises the obvious question as to why RES has concluded that its program is not cost-effective while other nations' regulatory authorities are actively pursuing this issue. We believe that it is of interest that none of these foreign programs are attempting to integrate organizational factors into PRAs.

It is our view that management science is a real and sophisticated academic field that needs to be tapped if the industry is to continue to make progress in dealing with organizational performance issues. There appears to be a lack of communication between the management science academic community and most policy-makers out in the "real world" of nuclear power plant regulation and operations. We believe that the Commission should encourage the involvement of the management science community in helping to improve the organizational performance of both the staff and the nuclear utilities.

RES Arguments for Terminating Organizational Factors Research - In SECY-93-020, RES makes the point that "the gathering of organizational factors data is resource intensive," but does not attempt to quantify this term. The presentations made to the Committee by the current contractors suggest that much less resource intensive approaches, relative to those used in the early phases of this work, are possible. The real test will be in the application of the products of this research when the benefits obtained can be compared to the resources invested.

RES also states that "there is a relatively low cost-effectiveness in continuing regulatory research beyond FY 1993, until it can be determined that organizational factors can be reliably integrated into PRA models." We were told by the contractors that the development and validation of these measurement tools are necessary before the integration of organizational factors into PRA models can be properly demonstrated. RES appears to have created a classic catch 22 situation in the position it has taken.

The Implications of Terminating Organizational Factors Research - RES states in the SECY paper that "the research products developed to date will be integrated by the end of FY 1993 for possible use in inspection and diagnostics evaluations." Based on our discussions with the contractors, we have concluded that the program to develop and verify organizational factors measurement tools is far from being completed. It appears to us that there is a major risk in exporting the present products to the field, since their almost certain unsuccessful application will bring this work into disrepute and create a significant obstacle to future developments in this field.

The Cost of Completing the Present Organizational Factors Research Program - The contractors were asked for their estimates of the time and cost to carry the present research to the point where a set of useful products (both organizational factors measurement tools and PRA modeling techniques) would become available for trial use by the staff and the nuclear utilities. They indicated that this would require an additional three years of effort at an annual funding of about $0.5 million (a small fraction of the current research program support budget). This additional $1.5 million expenditure is to be contrasted with the $3.8 million that has been expended on organizational factors research since 1987.

Our Reasons for Supporting Continuation of the Present Organizational Factors Research Effort - We believe that there is a reasonable expectation that products useful to both the NRC and the industry will be developed if the present program is completed. We further believe that completion of this program meets the benefit/cost test when compared with the expected benefits of many other research activities that have been, and are continuing to be, supported by the staff.

We see a strong analogy between the present status of organizational factors research and the status of PRA methodology 20 years ago when the Reactor Safety Study, WASH-1400, was begun under the leadership of the AEC. There were many, both within the NRC and industry, who argued at the time that PRA was a nice theoretical exercise, but would never have practical uses. Today, PRA is employed as an extremely valuable, multi-use tool by both the NRC and the regulated industry. Without this initial leadership by the agency, it is doubtful that PRA would be at today's state of development.

We believe that it is likely that the organizational factors measurement tools that are currently under development and their possible integration into PRAs will play an important role in nuclear power plant safety technology in the years to come. We do expect that it will be necessary, just as it was with the development of PRA, for the NRC and industry to expend additional resources on organizational factors research.

There are considerable demands presently being placed on staff and licensee resources in such activities as the SALP Program and Diagnostic Team Inspections. For licensees, the periodic INPO evaluations create additional demands. If appropriately validated organizational factors measurement tools can be developed, it would be possible to optimize the use of staff and licensee resources in assessing licensee organizational performance. The present staff approach in assessing licensee organizational performance does not have an appropriately validated basis and is subject to legal challenge (such a challenge has already been made with respect to the SALP Program). Continuing this research program to provide validated organizational factors measurement tools has the potential of providing the staff with a much more defensible basis for its SALP Program and Diagnostic Team Inspections.

After organizational factors measurement tools become available, it will be possible to undertake completion of the next step; the modeling of organizational factors into PRAs. If this modeling can be done in a credible manner, it would then be possible to assess how risk is apportioned between hardware and human performance. This would provide much needed insight into the manner in which NRC research efforts and inspection and enforcement resources should be allocated. It would also assist the staff and licensees in evaluating and correcting risk-significant weaknesses in their organizations.

We do not, however, believe that the integration of organizational factors into PRA should be the main focus of the present research program. Due to the complex, amorphous, and temporal nature of organizational performance, this objective may not be attainable. Rather, we believe that the emphasis should be on providing organizational effectiveness measurement tools to help the staff and the utilities better design and manage their organizations and to help the NRC make better judgments about the performance of licensee organizations. If the present integration efforts produce useful PRA input, so much the better. (We do believe that progress has been made by the researchers involved in this effort and recommend that this work be continued.)

Finally, we believe that the manner in which this research program has been carried out by the staff is representative of a serious generic problem that the staff has in dealing with complex issues that cut across staff organizational boundaries. We recommend that the EDO review the manner in which the various elements of the staff collaborated in developing the research objectives and in providing consistent guidance to the organizational factors research contractors. We expect such a review to lead to improved staff policy guidance on the coordination of future research efforts of this nature.

<u>Reference</u>:

SECY-93-020, dated February 1, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, Subject: Review of Organizational Factors Research

Revised: June 16, 1993

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

May 26, 1993

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

Dear Chairman Selin:

SUBJECT: STAFF APPROACH FOR ASSESSING THE CONSISTENCY OF THE PRESENT REGULATIONS WITH RESPECT TO THE COMMISSION'S SAFETY GOALS

During the 397th meeting of the Advisory Committee on Reactor Safeguards, May 13-15, 1993, we discussed a draft Commission paper regarding the staff's proposed approach for assessing the consistency of present regulations with respect to the Commission's safety goals. During this meeting, we had the benefit of discussions with representatives of the staff.

In a Staff Requirements Memorandum (SRM) dated June 15, 1990, the Commission requested that the staff develop a plan "for assessing the consistency of our regulations with the safety goals." This is an effort that the Committee has recommended in several reports, and continues to endorse.

In its presentation, the staff provided a conclusion that a specific new program is not necessary to respond to the SRM. The staff contends that existing programs, in the areas noted below, are sufficient to make the desired assessment:

1. Elimination of Requirements Marginal to Safety

2. IPE/IPEEE Data Base Insights

3. Other ongoing activities that include:

    - The Regulatory Review Group
    - Generic Safety Issue evaluations
    - AEOD evaluations of operational events and data
    - NRR inspection reports
    - Accident Sequence Precursor studies

We believe that these existing programs can provide input into the subject program, but are not by themselves responsive to the SRM. We recommend that a directed effort be undertaken to make the

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assessments requested in the SRM. A first step should be to develop an assessment strategy to make use of the IPE/IPEEE results and other appropriate PRA results to establish the existing level of safety that has resulted from compliance with the body of current regulations, to be compared with the safety goals.

The facts that the IPEs are essentially Level 2 PRAs and do not evaluate risk directly, and that seismic and fire events in IPEEEs are not necessarily evaluated probabilistically, are formidable barriers to their use for assessing the consistency of the present regulations with the safety goals. Nevertheless, these and other existing PRAs are the best available information for such an assessment. We recommend that the assessment strategy include the development of surrogates for the safety goals, expressed in terms of core damage probability and conditional containment failure probability – the outputs of the IPE. We believe that bounding, site-independent surrogates can be developed because, for high source terms, the conditional mean individual risk of early fatalities approaches a limit of about 0.1, and the conditional mean individual risk for latent fatalities approaches a limit of about 0.01. These limits result from the probability that the wind will blow in a given direction.

It is entirely possible that the outcome of such an assessment will reveal that the level of risk resulting from compliance with the body of existing regulations is below the safety goal levels of risk. Such a finding would have significant implications. It is important that such a determination be made.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Reference:

1. Memorandum dated April 18, 1993, from C. J. Heltemes, Office of Nuclear Regulatory Research, for John T. Larkins, ACRS, Subject: Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals, with attachments:
   a. SRM dated June 15, 1990, Subject: SECY-89-102, Subject: Implementation of the Safety Goals
   b. Draft SECY paper for the Commissioners from James M. Taylor, EDO, Subject: Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals (Predecisional)

2. ACRS Report dated April 12, 1988, from W. Kerr, ACRS Chairman, to The Honorable Lando W. Zech, Jr., NRC Chairman, Subject: Program to Implement the Safety Goal Policy -- ACRS Comments

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

July 15, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED DRAFT REGULATORY GUIDES, [illegible]-1023, "EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB," AND DG-1025, "CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE"

During the 399th meeting of the Advisory Committee on Reactor Safeguards, July 8-9, 1993, we discussed the subject draft regulatory guides. Our Subcommittee on Materials and Metallurgy examined these guides in detail at a meeting on June 29, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

The need for these proposed guides was highlighted during the evaluation of the integrity of the Yankee Rowe reactor pressure vessel. We believe that these guides should prove useful to the licensees and regulatory authorities, and recommend that they be issued for public comment. We would like an opportunity to review the proposed final version of these guides after the public comments have been reconciled and before they are published in final form.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

Reference:
Memorandum dated June 10, 1993, from Lawrence C. Shao, Office of Nuclear Regulatory Research, for John T. Larkins, ACRS, Subject:

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Request for ACRS Review of Proposed Draft Regulatory Guides, with enclosures:

- DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb," and
- DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

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

September 20, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE AMENDING FRACTURE TOUGHNESS REQUIREMENTS FOR LIGHT WATER REACTOR PRESSURE VESSELS, PROPOSED RULE REGARDING REQUIREMENTS FOR THERMAL ANNEALING OF REACTOR PRESSURE VESSELS, AND DRAFT REGULATORY GUIDE ON FORMAT AND CONTENT OF APPLICATION FOR APPROVAL FOR THERMAL ANNEALING OF REACTOR PRESSURE VESSELS

During the 401st meeting of the Advisory Committee on Reactor Safeguards, September 9-10, 1993, we discussed the subject proposed rules and draft regulatory guide. Our Subcommittee on Materials and Metallurgy reviewed these matters in detail at a meeting on August 16, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

The need for these proposed rules and the draft guide was, in part, highlighted during the evaluation of the integrity of the Yankee Nuclear Power Station's reactor pressure vessel. We believe these rules and this guide should prove useful to the licensees and the NRC staff and recommend that they be issued for public comment. We would like an opportunity to review the proposed final version of these rules and guide after the public comments have been reconciled and before publication.

Additional comments by ACRS Members Ivan Catton and William J. Lindblad are presented below.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

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Additional Comments By ACRS Members Ivan Catton and William J. Lindblad

Although we agree with the essence of the above letter, we oppose the elimination of the provision in Appendix H which currently permits a reduction of testing in Integrated Surveillance Programs where "initial results agree with predictions." The licensee's program is, after all, subject to staff approval on a "case-by-case basis." Licensees should have some flexibility in scheduling when they actually test specimens. This does not mean that specimens wou`: not be irradiated.

Refe. ce:

Memorandum dated August 20, 1993, from Allen L. Hiser, Jr., Office of Nuclear Regulatory Research, for Elpidio G. Igne, ACRS, Subject: Response to Request at ACRS Subcommittee Meeting, with the following:

a. Federal Register Notice for Proposed Rule, 10 CFR Part 50, "Fracture Toughness Requirements for Light Water Reactor Pressure Vessels"

b. A proposed rule (10 CFR 50.66) on thermal annealing of the reactor pressure vessel, "Requirements for Thermal Annealing of the Reactor Pressure Vessel"

c. Amendments to 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements"

d. Amendments to 10 CFR Part 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

e. A draft regulatory guide (DG-1027), "Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels"

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

November 16, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

On March 18, 1993, we wrote you a report on the NRC staff approach to regulation of computers in nuclear power plant operations and upgrades. While there were many specific observations and suggestions in that letter, it ended by concluding that a fresh start was called for in developing an effective approach to this new and difficult subject, and recommended that you ask the National Academies of Sciences and Engineering to conduct a workshop directed at this end.

In the interim the staff has conducted its own workshop on digital systems, with the help of the National Institute of Standards and Technology, on September 14-15, 1993. Some of us attended that workshop, and our Chairman gave introductory remarks. It is therefore appropriate to ask whether that workshop served as a reasonable substitute for our earlier recommendation. We have concluded that it did not.

To begin, it was not a workshop, in the usual sense of the word. It was organized much as a technical session of a learned society, with a variety of relatively disconnected speeches by experts, limited opportunity for questions from the audience, and only a little opportunity for the experts to discuss the issues with each other.

The recommendation in our earlier letter was based on the belief that an open-minded approach, using the wealth of expertise in the outside world, might help to supply the framework on which a coherent regulatory structure might be hung. Wrangling over specific details of the staff position, like the requirement for hard-wired redundancy, or concentration on electromagnetic interference, could lead to a compromise animal, half fish and half cat, with little underlying rationale.

Based on our observation of the staff workshop, and discussions with our foreign colleagues during the recent Quadripartite Meeting of Advisory Committees, we have concluded that our recommendation to seek help outside, with a different format, remains appropriate.

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The NRC can muddle through the next few years on current momentum, but lack of an underlying rationale will ultimately exact a price, perhaps a high one. There are deep issues of regulatory philosophy here, and a case-by-case approach will continue to ignore them.

We repeat our original recommendation.

Sincerely,

J. Ernest Wilkins Jr.

J. Ernest Wilkins, Jr.
Chairman

Reference:
Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

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

November 18, 1993

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

Dear Chairman Selin:

SUBJECT: NRC CONFIRMATORY TEST PROGRAM IN SUPPORT OF THE AP600 DESIGN CERTIFICATION

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we reviewed selected aspects of the NRC Office of Nuclear Regulatory Research (RES) experimental program to be conducted at the Japan Atomic Energy Research Institute's (JAERI's) Large-Scale Test Facility (LSTF) in support of the NRC design certification of the Westinghouse (W) AP600 passive plant. Our Subcommittee on Thermal Hydraulic Phenomena met on October 28, 1993, to review this matter. During this review, we had the benefit of discussion with representatives of the NRC staff. We also had the benefit of the documents referenced.

In a September 16, 1992 Staff Requirements Memorandum, the Commission requested that the ACRS review selected aspects of the ROSA-V test program prior to its initiation. Specifically, the Committee was asked to review the test matrix and the facility modifications and additions, including instrumentation and controls. The following comments are offered in response to that request:

- The modified LSTF has been designated as ROSA-V. Despite the modifications, a number of atypicalities and scaling distortions exist in the ROSA-V configuration relative to the AP600 design. Some of the atypicalities in ROSA-V are: the use of one cold-leg per reactor coolant system (RCS) loop instead of two; the geometry and heat transfer characteristics of the steam generators; the existence of a four foot loop seal in the RCS; excess metal mass (in particular, for the core makeup tank (CMT)); the volume and geometry of the in-containment refueling water storage tank (IRWST); the primary residual heat removal (PRHR) system; and the configuration of the pressurizer surge line. RES staff representatives stated that they understand the impact these atypicalities will have on system performance. The RES staff has not, however,

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presented a convincing argument that it understood the impact. RES should do so and document the results.

- Despite the facility shortcomings, we believe that ROSA-V will generate useful data to support validation of the relevant computer codes. This validation, however, may be inconclusive given the above atypicalities, especially those existing in the CMT, the PRHR system, and the IRWST. We recommend that the staff be urged to resolve the issues resulting from the atypicalities discussed above by additional analyses and, if necessary, by separate effects tests.

- The instrumentation proposed in support of the planned test program appears adequate for code assessment when dealing with single-phase phenomena. It is not clear that it is adequate for the measurement of key phenomena under conditions of two-phase flow. It is inadequate for determining some of the heat transfer characteristics of the PRHR system.

- The AP600 automatic depressurization system (ADS) will be activated by decreasing water level in the CMT. This level will be measured with heated junction thermocouples (HJTCs). The three AP600 integral system test facilities (ROSA-V, APEX—Advanced Plant Experiment—and SPES-II) will use differential pressure (DP) cells to measure this level. Activation of the ADS using DP cells rather than HJTCs could result in significant test distortions, given the inherent time delay associated with the use of HJTCs. The RES staff believes that these differences can be addressed. We were told by RES that JAERI has installed HJTCs of its own design at ROSA-V. We recommend that the RES staff use these HJTCs for ADS control for at least one properly chosen test, even if they are of a different design from those planned for use on the AP600.

- The ROSA-V test matrix is based on examination of transients and design-basis accidents for existing PWR designs. A number of the tests in the ROSA-V Phase I matrix have counterparts in the test matrices of the W SPES II and APEX facilities. These three facilities are scaled differently and have atypicalities of differing natures. We believe that the data obtained from these facilities will prove adequate for the necessary computer code validation by providing a broad range of challenges for simulation, given that the separate effects test programs supply sufficient information for code model development.

- Recently, RES modified the Phase I test matrix in response to a request from NRR to include some very small breaks and some "beyond-DBA" type events. We support this modification, but note that the capability of the relevant computer codes to model very small-break LOCAs is weak. This may lead to difficulties when code validation is attempted.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. NRC Report, NUREG/CR-6066 (Draft), "Analysis of LSTF Scaling for AP600 Testing," M. Ortiz, et al., June 11, 1993 (Draft Predecisional)
2. Memorandum dated December 23, 1992, from G. Rhee, NRC, to P. Boehnert, ACRS, transmitting INEL Report by T. Boucher, et al., "Description of Design Requirements for ROSA Modifications to Simulate AP600 Phenomena" (Revised September 1992)
3. U.S. NRC Report, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA Large-Scale Test Facility for AP600 Safety Assessment," M. G. Ortiz, et al., December 1992
4. ACRS report dated July 17, 1993, "Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification"
5. Staff Requirements Memorandum dated September 16, 1992, from S. J. Chilk, Office of the Secretary, to J. M. Taylor, EDO, "SECY-92-219 - NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design"
6. SECY-92-219, Memorandum dated June 16, 1992, from J. M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design

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

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.

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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.

J. Ernest Wilkins, Jr.
Chairman

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

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

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.

J. Ernest Wilkins, Jr.
Chairman

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