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FOR IMMEDIATE RELEASE
(Wednesday, November 4, 1992)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received from its independent Advisory Committee on Reactor Safeguards two letter-type reports. They provide comments on SECY-92-287, "Form and Content for a Design Certification Rule," and a second interim report on the use of the design acceptance criteria process in the certification of General Electric Nuclear Energy's advanced boiling water reactor design.

In addition, the ACRS has sent to the NRC's Executive Director for Operations three letter reports. They concern proposed guidance for implementation of the NRC's rule on maintenance at nuclear power plants, a proposed branch technical position on environmental qualification of electrical equipment for license renewal and proposed amendments to the NRC's Part 55 regulation on renewal of nuclear power plant operator licenses.

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Attachments:
As stated

October 16, 1992

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

Dear Chairman Selin:

SUBJECT: SECY-92-287, "FORM AND CONTENT FOR A DESIGN CERTIFICATION RULE"

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we reviewed SECY-92-287, "Form and Content for a Design Certification Rule." Our Subcommittee on Improved Light Water Reactors also discussed this matter on September 23, 1992. During these meetings, we had the benefit of discussions with representatives of the NRC. We also had the benefit of the documents referenced.

The staff is proposing the form and content for a design certification rule, in accordance with 10 CFR Part 52-Subpart B, which implements the intent of Part 52.

The staff proposes that any rulemaking changes to Tier 2 information requested by the NRC staff or a third party are to be governed by the backfit standard of 10 CFR 50.109(9)(a)(3). We note that this is inconsistent with previous Commission guidance given in the Staff Requirements Memorandum (SRM) regarding SECY-90-377 that "... the staff should be held to the backfitting standards of 10 CFR 52.63 for all matters resolved in the design certification rulemaking (in both tiers 1 and 2)." We recommend that the staff adhere to the Commission guidance in this regard, and apply the "adequate protection" standard to such changes.

The proposed rule would require the consolidation of all the design-related information into a single stand-alone document called the Design Control Document (DCD). The DCD would contain the two-tiered design-related information that would be extracted by the applicant from its application for design certification. Tier 1 includes the design information that is relied upon as the fundamental basis for the staff's safety review. It would include the design descriptions; inspections, tests, analyses, and acceptance criteria (ITAACs); site parameters; and interface requirements. Tier 2 is the remainder of the design-related information that is used in support of the certified standard design. The staff would review the DCD and provide its evaluation in a final safety evaluation report (FSER) for the design. The DCD would be referenced in the proposed standard design certification rule.

We recommend that you approve the staff's proposed form and content for a Part 52 Standard Design Certification Rule, subject to the following comments.

The staff does not propose to include its FSER as an integral part of the DCD. However, the FSER should be given clear standing for future interpretation of the rule, analogous to the manner in which a statement of consideration serves for other rules.

It is our opinion that, irrespective of the degree of care and effort applied to minimize the potential for ambiguities or inconsistencies, such problems will arise within the large volume of DCD material and with its evaluation in the FSER. Items not clarified in Tier 1 will have to be examined and settled on an ad hoc basis by consideration of intent at the Tier 2 level. Such an examination should include both the FSER and applicant documents.

On the question of secondary references, we propose that all documents and references that were considered important to the staff in making its final safety determination be identified in the FSER. These should be the only references to be designated as "resolved." Copies of those references that are not readily available should be included in the application for certification.

Sincerely,

David A. Ward, Chairman
Advisory Committee on
Reactor Safeguards

References:

1. SECY-92-287, dated August 18, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Form and Content for a Design Certification Rule
2. Memorandum dated February 15, 1991, from Samuel J. Chilk, Secretary of the Commission, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-90-377 - Requirements for Design Certification Under 10 CFR Part 52
3. Staff Requirements Memorandum M920908, dated September 30, 1992, from Samuel J. Chilk, Secretary of the Commission, for James M. Taylor, Executive Director for Operations, NRC, Subject: Form and Content for a Design Certification Rule and Follow-up to SECY-90-016 (SECY-92-287)

October 16, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission

Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: SECOND INTERIM REPORT ON THE USE OF THE DESIGN ACCEPTANCE CRITERIA PROCESS IN THE CERTIFICATION OF THE GENERAL ELECTRIC NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we continued our deliberations regarding the use of the design acceptance criteria (DAC) process and associated inspections, tests, analyses, and acceptance criteria (ITAAC) in the certification of the General Electric Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR) design. Our Ad Hoc Subcommittee on Design Acceptance Criteria considered this matter during its October 7, 1992 meeting. This Subcommittee was established to review the DAC process as requested by the Commission in its April 1, 1992, Staff Requirements Memorandum.

During these meetings we considered SECY-92-299, dated August 27, 1992, which is a staff status report on the subject of the development of DACs for the ABWR certification in the areas of instrumentation and controls (I&C) and control room design. It was evident from our meetings that the staff's review of these DACs and preparation of the supporting draft Final Safety Evaluation Report (FSER) chapters will require extensive further work. During these meetings, we had the benefit of discussions with representatives of the NRC staff and GE. We also had the benefit of the documents referenced.

Our first interim report on the DAC process, dated June 16, 1992, focused mainly on the other two DACs proposed by GE for use in certification of the ABWR design, namely, ITAAC 3.7 "Radiation Protection" and ITAAC 3.3 "Piping Design." We concluded that these DACs (with certain clarifications to the language of the drafts we reviewed) can provide an acceptable basis for the staff's final safety determination needed for design certification. We understand that these DACs will be available in final form for completing our review as part of the FSER. The staff is unable at this time to provide a schedule for completion of the FSER.

This interim report deals with the remaining two DACs — control room design, and instrumentation and controls. In our June 16, 1992 interim report, we indicated that these DACs had not been developed to a point where we could offer an opinion as to their acceptability. We did express concerns to the staff on several aspects of these DACs as they existed at that time. The staff has subsequently responded to these concerns.

Control Room Design DAC

Enclosure 3 of SECY-92-299 contains the DAC (i.e., ITAAC 3.6 "Human Factors Engineering") proposed by GE for the ABWR control room design (human factors aspects), a draft of the staff's FSER for Chapter 18 of the Standard Safety Analysis Report (SSAR), "Human Factors," and a Human Factors Review Model developed by the staff. The staff certification of control room design will be based on the design process described in this ITAAC. The implementation of the control room design process will be the responsibility of the combined operating license (COL) applicant or holder.

The draft FSER contains three open items in this DAC area, all involving documentation issues, that are being completed by GE and will then require the review and approval of the staff. These open items appear to be easily resolvable.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.6 since the issuance of SECY-92-299. It was this new material, which had not been completely reviewed by the staff, that we reviewed. Although we had a number of suggested language clarifications, we conclude that this ITAAC (with appropriate modification) will be able to provide an acceptable basis for the staff's final safety determination needed for design certification. We will complete our review of FSER Chapter 18 and this ITAAC when these documents become available in final form.

Instrumentation and Controls (I&C) ITAAC

Enclosure 2 of SECY-92-299 contains the ITAACs proposed by GE for ABWR I&C and a draft of the staff's FSER for Chapter 7 of the SSAR, "Instrumentation and Control Systems." The staff notes that GE will not have submitted complete design information in the I&C area prior to design certification because this is an area of rapidly changing technology. GE proposes the DAC material be included in the Tier 1 design as one system ITAAC (2.75 "Multiplexing") and three generic ITAACs (3.2 "Instrument Setpoint Methodology," 3.4 "Safety System Logic and Control," and 3.5 "Software Development"). The implementation of the design process described in the Software Development ITAAC would be the responsibility of the COL applicant or holder. Our review focused on the Software Development ITAAC which describes a design process as contrasted to a design.

The draft FSER includes five open items and 19 confirmatory items in the I&C area that are being completed by GE and will require the review and approval of the staff.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.5 since the issuance of SECY-92-299. It was this new material, that had not been reviewed by the staff, that we reviewed. We had a number of suggested clarifications to the language of this ITAAC. In addition, there are certain characteristics of software which, when specified at the beginning of the development process, make later assessment far easier. We believe that the staff and GE should include this concept in the Software Development ITAAC. We conclude that this ITAAC has the potential of providing an acceptable basis for the staff's final safety determination needed for design certification. We will continue our review as more information becomes available.

Finally, we are concerned about the significant number of post-design certification activities associated with these two DACs - control room design, and instrumentation and controls. The COL applicant or holder will be responsible for carrying out these activities. This will involve extensive future negotiations with the staff. It will also have the effect of diminishing the value of certified designs and seems to us to be contrary to the spirit of 10 CFR Part 52. We believe that the argument that these DACs represent areas of rapidly changing technology is being overplayed by both the staff and GE in justifying the extent to which the DAC process is being used.

We will keep you informed as our review of the DAC process in the certification of the GE ABWR design continues.

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

Sincerely,

David A. Ward
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I have a reservation about the Committee letter, for the specific issue of software certification. I have already taken (Reference 4) a more relaxed position than the Committee in the general area of DACs. That position reflects my view that we are dealing with a mature industry, not at all inexperienced in the design of modern reactors, and therefore requiring a different style of regulation than may have been the case in an earlier period. The most effective role of NRC is through oversight of the safety of the industry product, rather than on certification of each detail. The DAC process lends itself to this kind of regulation, but only in areas in which the staff itself has the experience and expertise necessary to assume this more global role. I hope that the staff will not inhibit the application of modern technology through excessive specificity, as exemplified by the analog backup controversy, on which the Committee has previously commented (Reference 6).

I have a separate nagging problem with the DAC process, as it is now being implemented, one which is exacerbated in this case. The staff is negotiating with the industry not only the potential applicants' programs for compliance with the (still unclear) acceptance criteria, but also the nature of the very requirements that the applicants will later have to meet. It is important to be very circumspect about the NRC's role in this process, lest NRC independence be compromised.

<u>References</u>:

1. SECY-92-299, dated August 27, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design
2. Staff Requirements Memorandum M920305A dated April 1, 1992, from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, Chairman, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992
3. GE Nuclear Energy, "Tier 1 Design Certification Material for the GE ABWR," dated June 1992
4. Report dated February 14, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews
5. Report dated June 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Interim Report on the Use of Design Acceptance Criteria in the Certification of the GE Nuclear Energy Advanced Boiling Water Reactor Design
6. Report dated September 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Digital Instrumentation and Control System Reliability

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October 15, 1992

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

Dear Mr. Taylor:

SUBJECT: PROPOSED GUIDANCE FOR IMPLEMENTATION OF THE MAINTENANCE RULE, 10 CFR 50.65

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we reviewed the NRC staff's proposed documents that provide guidance regarding implementation of the maintenance rule, 10 CFR 50.65. This rule is to become effective on July 10, 1996. Our Maintenance Practices and Procedures Subcommittee considered this matter during its October 6, 1992 meeting. During these meetings, we had the benefit of discussions with representatives of the NRC staff and NUMARC, and of the documents referenced.

The package of documents, which consists of a proposed regulatory guide and other supporting documentation, describes the staff proposal to endorse an industry consensus guidance document (Draft NUMARC 93-01) to implement the maintenance rule. The industry has a demonstration program in progress involving implementation of this guidance at nine nuclear power plants. The staff points out that its endorsement of this document maximizes "the leadership role of the industry in the area of maintenance." The staff believes that, "The performance based, results oriented characteristics of the maintenance rule make industry cooperation vital to successful implementation of the rule."

We agree with the staff's position and recommend that this package be issued for public comment.

We plan to review the staff's proposed final implementation guidance for the maintenance rule after the staff has resolved public comments, and to provide our comments to the Commission.

As presently proposed, the scope of the monitoring program with regard to the electrical connections to the utility transmission network is unclear. We recommend that the staff's final guidance be extended to include the switchyards.

During our meeting, we asked the staff to describe the progress it had made on developing guidance to the industry for implementing a maintenance program to satisfy the maintenance rule, and which also addresses the requirements of the license renewal rule. We had raised the issue of the need for such guidance in our August 17, 1992 letter to you on license renewal. Based on our discussions with the staff, we believe that continuing senior staff management attention to this issue is needed in the interest of coherence in the regulatory process. We also note that the reliability assurance programs being required of ALWR licensees will involve the establishment of a third kind of maintenance program. Consistent staff guidance is needed on the elements of an acceptable program that will satisfy these three sets of requirements.

Sincerely,

David A. Ward, Chairman
Advisory Committee on
Reactor Safeguards

References:
1. Memorandum dated September 9, 1992, from C. J. Heltemes, Jr., Office of Nuclear Regulatory Research, for Raymond F. Fraley, ACRS, Subject: Transmittal of a Proposed Public Comment Package Regarding Implementation Guidance for the Maintenance Rule, 10 CFR 50.65, with Enclosures
2. Memorandum dated May 5, 1992, from Jack W. Roe, Office of Nuclear Reactor Regulation, for Addressees, Subject: A Comparison of Maintenance and License Renewal Rules, with Enclosure
3. Draft NUMARC 93-01, Revision 2A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated July 1992

October 22, 1992

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

Dear Mr. Taylor:

SUBJECT: PROPOSED BRANCH TECHNICAL POSITION ON ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT FOR LICENSE RENEWAL

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we reviewed a proposed Branch Technical Position (BTP) on Environmental Qualification of Electrical Equipment for License Renewal. Our Subcommittees on Plant License Renewal and Reliability and Quality reviewed this matter during a joint meeting on September 16, 1992. The staff proposes that the BTP be issued for public comment. During these meetings, we had the benefit of discussions with members of the NRC staff, its consultants, and representatives of industry. We also had the benefit of the documents referenced.

Under the License Renewal Rule, 10 CFR Part 54, applicants will be required to develop a comprehensive program to identify in their plants all structures, systems, and components (SSCs) which may be subject to age-related degradation unique to the license renewal period. A further program to manage these components to ensure continued safe operation of the plant is also required.

The staff is now proposing an additional program, by means of a BTP, which singles out environmental qualification of electrical equipment for special treatment in the license renewal period. The particular concern of the staff seems to be that the qualification standards for insulation used on electrical cables prior to 1984 (representing 87 of 111 licensed nuclear power plant units) may not ensure adequate performance of cables for extended plant life. That, of course, is the issue for all SSCs in a plant, and it is not clear to us why the more general treatment of SSCs called for under 10 CFR Part 54 is not adequate for electrical cables as well.

Industry representatives expressed objection to the staff proposal for a BTP. They believe that while older plant cables were qualified to a lesser standard than has been in use since 1984, these cables have been approved for continued use in the plants (as has much other equipment where standards have evolved) and are part of the Current Licensing Basis (CLB) for each of these plants. Their interpretation of 10 CFR Part 54 is that the CLB is to be preserved with the exception that those SSCs subject to age-related degradation unique to the license renewal period should be subjected to specific management programs. They see no need for the BTP and believe it will result in unnecessary cable replacements and add significantly to plant costs for license renewal.

We are not convinced that the proposed BTP has been shown to be necessary or appropriate. It should not be issued for public comment until the matters discussed below have been addressed.

Neither the staff nor the industry presented any risk perspective on this issue. In simple terms, the risk is as follows: During the license renewal period the electrical cable in a key system might degrade in a way that the degradation would remain undetected during normal operation and by normal maintenance, testing, and surveillance practices. Then, during an accident, i.e., a LOCA, the insulation would fail and the key system would not perform its design function to mitigate effects of the accident. Present licensing practice assumes, and experience seems to confirm, that the probability of this sequence during the initial license period is acceptably low. At issue is whether the probability during the license renewal period is significantly greater. No evidence has been presented either way. Analysis of the risk importance of this issue should be made before the BTP is finally accepted or rejected. Such an analysis should include estimates of downside risks inherent in major projects intended to improve nuclear power plant safety.

Many electrical cables are covered with fire retardant materials. These coatings could have important effects on the aging of the cable insulation. Apparently, these effects have not been considered by the staff in development of this BTP. We do not know whether they have yet been explicitly considered in the selection and evaluation of important SSCs in license renewal programs. They should be.

Dr. Thomas Kress did not participate in the Committee's deliberations regarding this matter.

Sincerely,

David A. Ward, Chairman
Advisory Committe on
Reactor Safeguards

References:

1. Memorandum dated July 10, 1992, from John W. Craig, Office of Nuclear Reactor Regulation, NRC, for Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Branch Technical Position on Environmental Qualification of Electrical Equipment for License Renewal, with enclosures
2. Letter dated October 7, 1992, from M. H. Philips, Jr., and W. A. Horin, Counsel to the Nuclear Utility Group on Equipment Qualification, to D. A. Ward, Advisory Committee on Reactor Safeguards, Subject: NRC Staff Proposed License Renewal BTP Regarding Environmental Qualification of Electric Equipment, with enclosures

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October 19, 1992

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

Dear Mr. Taylor:

SUBJECT: PROPOSED AMENDMENTS TO 10 CFR PART 55 ON RENEWAL OF NUCLEAR POWER PLANT OPERATOR LICENSES AND REQUALIFICATION

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we reviewed the proposed amendments to 10 CFR Part 55. During this meeting, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

These proposed amendments would revise the current requalification regulations for licensed operators at nuclear power plants by eliminating the present requirements that they pass a requalification written examination and operating test administered by the NRC during their six-year license term.

Licensed operators would continue to be required to pass the biennial requalification written examination and annual operating test administered by their plant training organizations. As part of the proposed rule change, licensees would be required to submit their examinations and operating tests for NRC review. The staff points out that these changes in the regulations will allow the redirection of NRC license examiner resources so that the examiners will be able to perform more comprehensive, programmatic inspections of licensee operator training programs.

We believe that these proposed amendments to 10 CFR Part 55 will be beneficial and recommend that they be released for public comment. We would like the opportunity to review the proposed final version of these amendments after the staff has reconciled the public comments.

Sincerely,

David A. Ward, Chairman
Advisory Committee on
Reactor Safeguards

References:

Memorandum dated September 11, 1992, from C. J. Heltemes, Office of Nuclear Regulatory Research, NRC, for Raymond F. Fraley, ACRS, Subject: Request for Review of Proposed Rule Change to 10 CFR Part 55 and Associated Regulatory

Analysis, with Enclosure 1, Commission Paper on Proposed Amendments to 10 CFR Part 55, and Enclosure 2, Status and Direction of the Licensed Operator Requalification Program, SECY-92-100, March 19, 1992