No. 92-96 FOR IMMEDIATE RELEASE
Tel. 301/504-2240 (Thursday, June 25, 1992)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received four letter-type reports from its independent Advisory Committee on Reactor Safeguards. They provide comments on:

1) implementation of the Commission's safety goal policy for nuclear power plants;

2) the use of design acceptance criteria in the certification of the General Electric Company advanced boiling water reactor design;

3) testing and analysis programs in support of GE's simplified boiling water reactor design certification; and

4) proposed amendments to the NRC's fitness-for-duty rule.

In addition, the NRC's Executive Director for Operations has received two letter reports that comment on individual plant examination and accident management programs and two Regulatory Guides related to implementation of the NRC's revised Part 20, "Standards for Protection Against Radiation."

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Attachments:
As stated

June 12, 1992

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

Dear Chairman Selin:

SUBJECT: IMPLEMENTATION OF THE SAFETY GOAL POLICY

In our report of December 18, 1991, we expressed reservations about a staff proposal contained in SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," and offered to produce an alternative approach after we had further considered the matter. You urged us to do so. During our 386th meeting, June 4-5, 1992, we developed the following proposal for consideration.

We assume that the Commission desires to establish guidance for the NRC staff to ensure that regulatory activities will be conducted in a manner consistent with the intent expressed in the Safety Goal Policy Statement. Although the plan outlined in SECY-91-270 purports to provide such guidance, we have major reservations about it. First, it applies only to a small part of the spectrum of regulatory activities and should not be characterized as <u>the</u> plan to implement the policy. A more strategic vision for implementation is needed. Second, even as a tactical tool, part of an overall implementation program, the SECY-91-270 plan has some significant inconsistencies with the policy.

We interpret the safety goals to be an expression of "how safe is safe enough." Thus, the Policy Statement expresses the Commission's intention that the safety of the general population of plants should be consistent with the goals, but implies no requirement or expectation that either individual plants or the population of plants must surpass the goals. The Safety Goal Policy Statement defines an acceptable level of safety for the nuclear enterprise.

This means that regulatory programs should not be an unending quest for higher and higher nominal levels of safety, but should be directed instead toward providing assurance that the plants, as a whole, meet the standard of safety already proclaimed by the Commission.

COMMISSION USE OF THE POLICY

The most important use of the safety goals should be by the Commission itself. The goals describe the level of safety which the Commission promises to achieve through its regulatory efforts. The body of regulatory activity in place is, in effect, its current effort to implement the safety goal policy. Whether this present activity is adequate can be partially evaluated by comparing the fruit of these practices, the safety level of operating plants, to the goals.

Data on safety performance are available from two sources. The first is the set of "bottom-line" risk estimates from the many available PRAs, including those described in NUREG-1150 and those being developed under the Individual Plant Examination (IPE) program. Imperfections in this body of data are clear and should be recognized, but the data contain important information. One defect is incompleteness, e.g., while the PRAs characterize the design and physical status of plants fairly well, they say little about how well the plant is operated.

The second source is the risk information deduced through the Accident Sequence Precursor (ASP) program from operating experience. These data are also imperfect and must be judiciously used, but they also contain important information.

An indication from present PRAs is that the general population of plants operates near the standard set by the safety goals. One might conclude that the present level of regulation is sufficient. But uncertainties abound.

Assessment of the ASP data is also difficult. Improvements in ASP methods are needed before solid conclusions can be drawn from them.

No general conclusion is apparent at this time. We are not sure any can be drawn beyond a general impression that there are no indications that U.S. plants are failing to operate in accordance with the Commission's goals. However, we believe these (PRA and ASP) are important attempts, really the only quantitative attempts, to evaluate the overall safety performance of U.S. nuclear power plants and, presumably, the effectiveness of the Commission's present efforts to implement the safety goal policy. One cannot know whether changes in regulations are necessary and sufficient unless one has some measure of how effective the body of regulations has been in fostering a population of plants which operates in accordance with the safety goals.

STAFF USE OF THE POLICY

Although we consider the strategic use of the safety goals, as discussed above, to be the most important, it is also appropriate, when feasible, that the policy serve tactical purposes. It is a mistake to expect that a tactical use can be easily correlated with the high-level safety goals. For example, in a narrow sense the Backfit Rule (10 CFR 50.109) could be considered to be in conflict with the goals, if the latter are accepted as a statement of "how safe is safe enough." Also, one inevitably confronts the uneasy relationship between the safety goals and the definition of adequate protection. Rather, the principal use of the safety goals in support of tactical decision-making tools by the staff should be to help ensure conformance with the policy.

One tactical use would be to evaluate the need for proposed enhancements to regulations. Strictly speaking, we don't have much basis, from the perspective of PRA and ASP studies carried out to date, to argue that enhancements are needed. However, the assessment of the safety of operating plants is an immature and incomplete undertaking. Until such assessments are further along, proposals for regulatory change, based on judgments about safety or on better understanding or knowledge, will evolve.

Given a particular contemplated regulatory action, a procedure can be developed to use surrogate guidelines, derived to be consistent with the safety goals, to decide its advisability. However, there should be no expectation that unambiguous yes-or-no answers can be established with such a procedure, simply by comparing risk estimates to the surrogate guidelines. Allowance for uncertainty and for unquantifiable factors must be included.

There are many proposals for additional requirements. These include resolution of matters typically brought before the Committee to Review Generic Requirements (CRGR). There is also systematic activity to help decide whether development of new requirements should be undertaken, e.g., the Generic Safety Issues (GSI) "prioritization" program. The proposal in SECY-91-270 was developed to evaluate the first of these. We understand that a similar proposal is being developed to provide guidance as to whether the initial development of new requirements should be undertaken. Neither of these proposals is inappropriate in concept; they could help ensure that these important staff activities reflect the intent of the safety goal policy. However, as we stated in our report of December 18, 1991, much of the SECY-91-270 plan seems to miss the full intent of the policy.

There are not many current proposals for deletion of requirements. This is unfortunate. There have been programs directed to this end, but they seem to have come up dry.

In our report dated May 13, 1987, on the Safety Goal Policy Statement, we commented on the use of surrogates to facilitate application of the safety goals to lower-level regulatory problems. In that report, we emphasized the importance of structuring the surrogates so that they remain surrogates, and do not become new de facto safety goals, more conservative than the original ones. While we emphasized the avoidance of excessive conservatism (because it is more of a problem at NRC), there are also pitfalls to be avoided in the other direction. It is Scylla and Charybdis, and one requires precise navigation to avoid the perils on either side.

But precise navigation is not possible in the world of PRA, the necessary tool for implementation of the safety goals. Any calculation of a probability has inevitably bound with it an uncertainty, and the uncertainty, however expressed, is as much a result of the analysis as the central number. We said more about this issue in our report dated December 14, 1991, on the use of PRA by the staff. (The undeniable allure of a precise decision mechanism leads all too often to staff decision making based on single bottom-line probability estimates, with at best lip service paid to uncertainty. We need to do better here.) While surrogate measures of risk at lower levels of aggregation can be invented and can be expressed as precise numbers, their tactical use must reflect the uncertainties associated with their calculation. It is important to distinguish between the statement of a surrogate (or indeed a goal) as a precise number, and its calculation for some given situation, which will contain uncertainty. Without this distinction, irrational decisions are not only possible, but are certain. If a calculation which is uncertain by a factor of 10 shows that a proposed rule change exceeds some threshold criterion by 10 percent, is it rational to implement the change?

Given the first caveat about avoidance of added conservatism through surrogation, various lower-level surrogates for the safety goals have been suggested over the years, applicable in different situations. Among them are probabilities of 1E-6 per reactor-year for a large release, 1E-4 per reactor-year for significant core damage, and 1E-1 for a conditional containment failure probability. Each of these deserves continuing consideration to provide assurance that it is neither unduly conservative nor the converse, each can be calculated with substantial (and quantifiable) uncertainty, and each seems to us a reasonable step toward a useful surrogate for the full safety goals in a regulatory decision-making process.

What is still needed is a means for incorporating the necessary uncertainties in the calculations into the decision making. There will be some cases for which the calculated effect of a proposed change will be so clearly above or below the requirements of the surrogate standard (taking the uncertainty into account) that the decision process is simple and beyond reasonable disagreement. This might be judged by choosing some appropriate statistically described confidence level for the ordering of the surrogate standard and the calculated effect of the proposed change.

The difficult problems appear when there is inadequate statistical confidence that the proposed change meets the threshold surrogate standard; presumably this will be far from an uncommon event. For such situations we can only say that there is no free lunch—if the probabilistic situation is uncertain, other criteria will have to be used to bring the matter to a conclusion. This is not so strange; before the probabilistic era began, these other criteria were all that was used in decision making. In this proposal they would be used only to augment the safety-goal-based considerations, and then only in the event of substantial uncertainty.

What are these other criteria? Apart from shibboleths like engineering judgment, they include optimization of effort, resource allocation, discounting of impact timing, the intangible safety benefits of stability (if it ain't broke, don't fix it), the number of plants affected (there may well be times when a proposed change to a few plants will appear desirable from the point of view of the safety of those plants, but they are so few that the impact on the public risk will be small), and a host of other considerations. We leave their invention to the Commission and the staff, and wish only to note that there are times when decision making is difficult.

We do note, as a matter of principle, that there is no probability that cannot be quantified—the only issue is the level of uncertainty associated with the quantification. By the same token, there is no usefulness to a calculated probability without an associated statement, in some quantitative form, of its uncertainty.

We think the scheme we have outlined above is workable, though we recognize that we have provided only its skeleton. We also recognize that there will have to be a learning phase, in which the staff subjects proposed enhancements (and the converse) to the kind of analysis described here. Indeed, there will have to be a learning process for the more strategic implementation of the safety goals that we described in the first part of the letter. If the Commission subscribes to this general approach, we will be happy to work with both you and the staff to bring this long enterprise to a constructive conclusion. At best, it can offer a structure for more efficient use of NRC resources, and more effective regulation of industry, by focusing attention on regulatory activities calculated, however imperfectly, to have the most impact on the health and safety of the public.

Additional comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins are presented below.

Sincerely,

David A. Ward
Chairman

Additional Comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins

Although we thoroughly approve of the direction proposed in this report, we regret that the Committee has chosen to make the reference to confidence levels on page 4 of the report so terse that the recommendation conceals real problems.

The suggestion that a decision-making mechanism can be based on a confidence level is workable, but the choice of a specific level—90% or 95% or whatever—involves a balance of benefits and effort that can only be resolved by the Commission. It is not a matter for fiat, but for analysis.

This is especially difficult for two reasons. The simplest one is the fact that the probability distributions in most PRAs are non-gaussian, so the familiar translation into a sigma level is inappropriate. That makes the application of a confidence criterion less straightforward than might appear on the surface, and departs from many engineers' experience.

Far more important, and ignored by the Committee, is the fact that the words "confidence level" mean different things to different people. Engineers tend to have little education in the subtleties, classical statisticians have little experience with low-probability analyses, and classical and Bayesian (we would say modern) statisticians both use the term "confidence level," but mean entirely different things by the term. These are not just semantic differences—they need to be resolved if a confidence criterion is to be used, lest the ambiguities render a difficult job impossible.

References:

1. SECY-91-270 dated August 27, 1991, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy
2. Staff Requirements Memorandum dated February 21, 1992, from Samuel J. Chilk, Secretary, for James M. Taylor, NRC

Executive Director for Operations, William C. Parler, General Counsel, and David A. Ward, Chairman, ACRS, Subject: SECY-91-270 - Implementation of the Safety Goal Policy Statement

3. Reports by the Advisory Committee on Reactor Safeguards on implementation of the safety goal policy:
   a. ACRS Comments on an Implementation Plan for the Safety Goal Policy, dated May 13, 1987
   b. SECY-91-270, Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy, dated December 18, 1991
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990

June 16, 1992

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

Dear Chairman Selin:

SUBJECT: INTERIM REPORT ON THE USE OF DESIGN ACCEPTANCE CRITERIA IN THE CERTIFICATION OF THE GE NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 386th meeting of the Advisory Committee on Reactor Safeguards, June 4-5, 1992, we continued our discussions with the NRC staff on the use of the design acceptance criteria (DAC) process in the certification of the GE Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR) design. Our Ad Hoc Subcommittee on DAC, which was established to review the DAC process as requested by the Commission in the April 1, 1992 Staff Requirements Memorandum, met with representatives of the NRC staff and GE on May 6, 1992, and with the NRC staff on June 3, 1992, to discuss this matter. We also had the benefit of the documents referenced.

In general, we are satisfied with the progress that the staff and GE are making in the development of the four DACs (each consisting of a set of DAC/ITAACs - Inspections, Tests, Analyses, and Acceptance Criteria) presently envisioned for use in the certification of the GE ABWR design. (The staff has indicated that it expects these same four DACs, with some modification in scope, will be used in the certification of the ABB-CE System 80+ design.) The staff views these DAC/ITAACs as a form of Inspections, Tests, Analyses, and Acceptance Criteria that commits the Combined Operating License (COL) holder to a design process with appropriate acceptance criteria that would be applied at various milestones during the design process (as contrasted to the normal ITAACs that will be used to confirm that a certified design has been constructed and tested by the COL holder in accordance with design commitments). These DACs are intended to provide the necessary and sufficient commitments on design processes that will be employed by the COL holder in implementing Tier 1 functional design requirements. These functional design requirements will be a part of the design certification.

The four DACs proposed by GE for use in the certification of the ABWR design appear to be consistent with the recommendations in our February 14, 1992 report to the Commission regarding the use of the DAC process. We note that each of these DACs poses different problems in specifying "practical and technically unambiguous acceptance criteria" in the absence of detailed design information. Because of this, it is our intent to review each of these DACs in detail in order to obtain a fuller understanding of the issues presented by the DAC process. Our final comments and recommendations will be made following our receipt and review of the individual Commission papers that the staff plans to prepare on these four DACs.

Based on our review to date, we have the following comments on the ABWR DACs currently under consideration:

Radiation Protection

- The radiation protection DAC, which the staff believes to be near completion, deals with the adequacy of the ABWR's radiation shielding, ventilation systems for airborne radioactivity areas, and airborne radioactivity monitoring systems. These DAC/ITAACs represent a subset of the staff's overall review of the ABWR radiation protection design features which, in aggregate, are intended to maintain radiation exposures for both plant personnel and the general public well below acceptable limits. GE's position is that it is not possible at this stage in the design to provide the level of detail specified in the applicable regulations, regulatory guidance, and the Standard Review Plan (SRP) for these three aspects of the ABWR's radiation protection design.
"As procured" information is not yet available for components that (1) will be radiation sources and will require shielding or (2) will be potential sources of leakage of radioactive fluids which will establish ventilation system requirements to limit concentrations of airborne radioactivity and the basis for suitable continuous airborne radioactivity monitoring systems. GE has proposed DAC/ITAACs to deal with these issues. These DAC/ITAACs would require the COL holder, following procurement of these components, to perform analyses to verify the adequacy of the plant shielding and the ventilation system design in airborne radioactivity areas and to identify those plant areas requiring continuous monitoring of airborne radioactivity and to provide appropriate monitoring systems. In each case, the DAC/ITAACs provide acceptance criteria that the staff believes are consistent with applicable regulatory requirements, regulatory guidance, and the SRP. The staff believes that compliance with this version of these DAC/ITAACs is acceptable as a basis for design certification pending its review and acceptance of the final version of these DAC/ITAACs.

We discussed these DAC/ITAACs (Tables 3.7a and 3.7b of GE's March 1992 Tier 1 Certification Material) and the staff's draft Safety Evaluation Report (SER) with the NRC staff during our meetings and suggested a number of clarifications to the language of these DAC/ITAACs. We believe that these DAC/ITAACs (with appropriate modification) can provide an acceptable basis for the staff's final safety determination needed for design certification.

Piping Systems

- The piping systems DAC, which the staff also believes to be near completion, deals with code-related design and analysis of ABWR piping systems important to safety.

  We note that the staff and GE agree that the analysis of such piping-related issues as flooding and compartment pressurization resulting from pipe breaks and the environmental effects of pipe breaks on other equipment in the vicinity of the break need to be explicitly included in the Tier 1 certified design commitments and their associated ITAACs. (The issue of flooding is already included as a Tier 1 commitment.) As such, these issues will not be a part of the piping systems DAC. We were also told by the staff that it will use the SRP as a basis for making its safety determination on these issues.

  We have two concerns regarding these issues. First, we continue to have difficulty envisioning how the staff will be able to make a final safety determination on the issues of compartment pressurization and the environmental effects of pipe breaks without having additional information on piping and equipment layouts beyond that presently available in the GE Standard Safety Analysis Report (SSAR). Secondly, we are concerned that these important issues remain to be resolved at this late date in the design certification schedule.

  With respect to the code-related piping systems DAC, GE's position is that it is not possible at this stage in the design to provide the level of detail specified in the applicable regulatory requirements because "as procured" information needed for piping analysis is not yet available for components (such as valves, pressure vessels and heat exchangers) that will be a part of these piping systems. GE has proposed DAC/ITAACs to deal with this issue. These DAC/ITAACs would require the COL holder, following procurement of components, to perform analyses of agreed-upon piping systems to show that the design meets the certified design commitments. In each case, the DAC/ITAACs provide acceptance criteria that the staff believes are consistent with applicable regulatory requirements, including conformance with ASME Section III, regulatory guidance, and the SRP. The staff believes that compliance with this version of these DAC/ITAACs is acceptable as a basis for design certification pending its review and acceptance of the final version of these DAC/ITAACs.

We discussed these DAC/ITAACs (Table 3.5 of GE's March 1992 Tier 1 Certification Material) and the staff's draft SER dated May 1, 1992, with the representatives of the NRC staff and GE during our meetings. As a result of these discussions, we suggested a number of clarifications to the language of these DAC/ITAACs. We believe that these DAC/ITAACs (with appropriate modification) can provide an acceptable basis for the staff's final safety determination needed for design certification on the issue of code-related design and analysis of ABWR piping systems important to safety.

Man/Machine Interface

- The man/machine interface (MMI) DAC deals with the implementation of a systematic approach to the incorporation of human factors principles in the detailed design of operator workstations in the control room and at the remote shutdown panel. Unlike the two DACs discussed above, this set of DAC/ITAACs has not been developed to a point where we can offer an opinion as to its acceptability as a basis for the staff's final safety determination needed for design certification.

  We did express a concern to the staff regarding the minimum inventory of fixed alarms, displays and controls that is being developed as a part of the Tier 1 design certification. Operator actions shown to be important based on the ABWR PRA are one basis for this inventory. In our letter of April 13, 1992 to the EDO we indicated that the ABWR PRA appeared to have a number of shortcomings. It is not clear when, or if, GE will redress these PRA shortcomings. This leads to the possibility that misleading information could be used in making control room MMI design decisions.

  We also expressed two concerns to the staff regarding the scope of these DAC/ITAACs under development: (1) these DAC/ITAACs should include the influence of transmission switchyard workstations, because of the importance of offsite power to the safety of nuclear power plant operation, and (2) the scope of these DAC/ITAACs should be expanded to include the incorporation of human factors principles in the design of local panels where instrumentation and controls important to safety are located.

Control and Protection Systems

- The control and protection systems design (I&C) DAC deals with the implementation of digital system designs to meet the functional specifications for those systems that will be established as part of the Tier 1 certification. Again, this set of DAC/ITAACs has not been developed to a point where we can offer an opinion as to its acceptability as a basis for the staff's final safety determination.

  We expressed a concern to the staff regarding the scope of these DAC/ITAACs in that they do not appear to include criteria for instrumentation and control systems hardware or hardware/software integration. The staff believes that these issues will be covered by the formal verification and validation program for the safety-related portions of the system and will clarify this point in a future revision of the Tier 1 material.

Sincerely,

David A. Ward
Chairman

References:

1. SECY-92-196, dated May 28, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)
2. Staff Requirements Memorandum (SRM-M920305A) dated April 1, 1992, Subject: Staff Requirements Periodic Meeting with the Advisory Committee on Reactor Safeguards
3. GE Nuclear Energy, Stage 2 Submittal, regarding Tier 1 Design Certification Material for the GE ABWR Design, dated March 30, 1992
4. Report dated February 14, 1992, from David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, to Ivan Selin, Chairman, NRC, Subject: Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews
5. Letter dated April 13, 1992, from David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, to James M. Taylor, Executive Director for Operations, Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design

June 10, 1992

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

Dear Chairman Selin:

SUBJECT: TESTING AND ANALYSIS PROGRAMS IN SUPPORT OF THE SIMPLIFIED BOILING WATER REACTOR DESIGN CERTIFICATION

During the 385th and 386th meetings of the Advisory Committee on Reactor Safeguards, May 6-9 and June 4-5, 1992, we reviewed the testing and analysis programs in progress and proposed by GE Nuclear Energy (GE) in support of the certification effort for the Simplified Boiling Water Reactor (SBWR) passive plant design. Our Subcommittee on Thermal Hydraulic Phenomena held meetings to discuss this topic on April 23 and June 2, 1992. During these meetings, we had the benefit of discussions with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

GE will use its best-estimate code, TRACG, to evaluate the SBWR thermal hydraulic behavior under accident conditions ranging from ATWS with instabilities to long-term behavior of the Passive Containment Cooling System (PCCS). GE representatives presented a very good analysis of processes and phenomena important to accident scenarios postulated for the SBWR. The results were summarized in tables which are to be used by GE to validate the TRACG computer code. However, these same tables appear not to have been used to guide the design and operation of the experimental facilities that are to support the code validation process.

The GE experimental program consists of three elements:

1) Laboratory scale experiments to obtain fundamental heat transfer data,

2) Separate effects tests to obtain data for parts of the total system and full-scale components where necessary, and

3) Integral system tests to obtain system data.

Although we were shown some comparisons of TRACG predictions with data from GE's integral system tests (GIST and GIRAFFE facilities), the question of whether or not the facilities can scale the important phenomena was not addressed in either GE's presentation or in the documents supplied to the ACRS by GE. A rigorous scaling analysis is needed if integral system test data alone are to be used to demonstrate that a TRACG calculation is meaningful.

We have some comments about the elements of the GE test plan. The initial conditions for the integral system tests are based on conditions assumed to exist some time after vessel depressurization. These conditions include an initial drywell and PCCS nitrogen mass fraction of 15 percent. The nitrogen concentration could be much higher. GE should develop a basis for its choices of initial conditions or broaden its test matrix to include some tests at much higher values of the nitrogen concentration, both in the drywell and in the PCCS.

Separate effects tests to be conducted in the PANTHERS facility will yield the data needed to characterize heat exchanger behavior under a variety of expected conditions. In particular, GE has agreed to add instrumentation to the individual heat exchanger tubes to obtain local heat transfer data. This will make the GIRAFFE integral system experiments more meaningful. We believe GE has been very responsive to issues raised by both the ACRS and the NRC staff in this regard.

The oscillatory behavior observed in the GIRAFFE integral system tests needs more detailed study to ensure that the suppression pool does not overheat due to steam bypass of the PCCS through the suppression pool top horizontal vents. The steam flow rate will be low which could lead to a stratified condition. The suppression pool is not a very effective heat sink when this process occurs. This may well require a separate effects study to obtain data for development of a low steam flow model for the horizontal vent. Further, review of the GIRAFFE facility instrumentation is needed to ensure that the resulting data will support TRACG model validation.

The SBWR has full pressure isolation condensers (IC) capable of removing 4.5 percent of full power decay heat at full system pressure. The behavior of isolation condensers is well understood and introduces no new processes. GE has indicated that it will collect relevant IC operating data for staff review. The SBWR is automatically depressurized when the vessel water level drops to some prescribed value by a staged opening of squib-type valves. Further, GE has had a great deal of experience with automatic depressurization and only the squib-type valve itself is of a new design. As a result, we do not believe that full-height, full-pressure integral system testing is required for certification of the SBWR design.

The GE program includes conduct of integral system testing at the PANDA facility located in Switzerland. The NRC staff would like GE to obtain data from this facility in time to support its design certification review of the SBWR. To do so, GE would have to accelerate its schedule by six months. We agree with the NRC staff that further integral system testing of the PCCS is needed prior to the final design approval. It has not been demonstrated by GE that existing data obtained from GIRAFFE or GIST testing are sufficient for validation of the TRACG code, nor that the PANDA test facility will yield the needed data. A more definitive assessment by GE is needed; this assessment should include both the scaling rationale for the GIRAFFE, GIST, and PANDA facilities, and a demonstration of how the effects of test facility scaling distortion impact the important processes and phenomena outlined by GE in its evaluation of TRACG. As a part of such an effort, it may be possible to show that one can obtain the needed data by some combination of additional separate effects tests and judicious use of the GIRAFFE and GIST facilities.

To summarize, we agree with the NRC staff views that full-height, full-pressure integral system testing is not needed to support the SBWR design certification. Further, we agree that early integral system testing of the PCCS is essential to meet the present design certification schedule. We have not, however, seen evidence that the PANDA facility is adequate to obtain the needed data.

Sincerely,

David A. Ward
Chairman

References:

1. Memorandum dated February 26, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, transmitting Advance Copy of proposed Commission paper, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)"

2. Letter dated February 3, 1992, from R. C. Mitchell, GE Nuclear Energy, to U.S. Nuclear Regulatory Commission, Subject: GE Response to Request for Information on SBWR Testing Program

3. Joint Study Report, "Feature Technology of Simplified BWR (Phase I) GIRAFFE (Final Report)," dated November 1990, The Japan Atomic Power Company, et al. (**GE Proprietary Information**)

4. GE Nuclear Energy, GEFR-00850, "Simplified Boiling Water Reactor (SBWR) Program Gravity-Driven Cooling System (GDCS)

Integrated Systems Test - Final Report," A.F. Billig, dated October 1989 (**Applied Technology Restriction**)

5. "ALPHA – The Long Term Passive Decay Heat Removal and Aerosol Retention Program at the Paul Scherrer Institute, Switzerland," by P. Coddington, et al., Paul Scherrer Institute, undated

6. Paper from the Proceedings of The International Conference on Multiphase Flows '91 – Tsukuba, Japan, September 24-27, "Condensation in a Natural Circulation Loop with Noncondensable Gases Part 1 – Heat Transfer," K. M. Vierow, GE Nuclear Energy, and V. Schrock, University of California

7. GE Draft Report: "Test Specification for IC & PCC Tests," undated (**GE Proprietary Information**)

8. Paper submitted to the Department of Energy, "The Effect of Noncondensable Gases on Steam Condensation Under Forced Convection Conditions," M. Siddique, Ph.D. Thesis - Massachusetts Institute of Technology, dated January 1992

June 10, 1992

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

Dear Chairman Selin:

SUBJECT: PROPOSED AMENDMENTS TO FITNESS-FOR-DUTY RULE
(10 CFR PART 26)

During the 386th meeting of the Advisory Committee on Reactor Safeguards, June 4-5, 1992, we reviewed the proposed amendments to the Fitness-For-Duty (FFD) rule which the staff plans to issue for public comment in the near future. Our review was in response to the November 7, 1991 Staff Requirements Memorandum (SRM), "SECY-91-293 – Assessment of Implementation of the Fitness-For-Duty (FFD) Rule and Need for Changes to the Rule." In that SRM, you asked us to review and comment on these proposed amendments prior to their submittal to the Commission. During our meeting, we had the benefit of discussions with representatives of the NRC staff and NUMARC. (NUMARC had not seen this proposed amendment package and was unable to provide detailed comments.) We also had the benefit of the documents referenced.

The staff has evaluated information from a number of sources to determine the effectiveness of the rule since it was implemented during January 1990, and to identify potential desirable changes to the rule. These evaluations included the results of inspections of licensee FFD programs, periodic licensee reports on program results, initiatives by the National Institute on Drug Abuse and NUMARC, Commission's instructions included in the SRM dated November 7, 1991, and a letter from NUMARC dated April 17, 1991 which provided some 51 proposed modifications to the rule based on the nuclear power industry's first year's experience with the FFD programs. We have no objection to the publication of the proposed amendments for public comment.

As an additional matter, because the existing testing program is resource intensive and there appears to be minimal use of drugs and alcohol at nuclear power plants, the staff proposes to investigate the possibility that a reduced testing schedule would be effective in maintaining an appropriate program for detection of substance abuse. We encourage this activity.

Sincerely,

David A. Ward
Chairman

References:

1. Staff Requirements Memorandum dated November 7, 1991 from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-91-293 – Assessment of Implementation of the Fitness-for-Duty (FFD) Rule and Need for Changes to the Rule
2. SECY-91-293 dated September 17, 1991 for The Commissioners from James M. Taylor, Executive Director for Operations, Subject: Assessment of Implementation of the Fitness-For-Duty (FFD) Rule and Need for Changes to the Rule
3. U.S. Nuclear Regulatory Commission, Draft Part 26 – Fitness for Duty Programs, dated April 20, 1992 (PREDECISIONAL)n

June 11, 1992

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

Dear Mr. Taylor:

SUBJECT: INDIVIDUAL PLANT EXAMINATION AND ACCIDENT MANAGEMENT PROGRAMS

During the 386th meeting of the Advisory Committee on Reactor Safeguards, June 4-5, 1992, we discussed the status of the Individual Plant Examination (IPE) and Accident Management Programs. This matter was also discussed with the staff and NUMARC during a joint meeting of our Subcommittees on Individual Plant Examinations and Severe Accidents held on April 21, 1992. We also had the benefit of the documents referenced.

The IPE program is achieving the objectives that we had hoped it would. The Accident Management Program was perceived to be at a reasonable, albeit an early, stage of development. However, the staff and NUMARC appear to be developing, cooperatively, a program that will extend beyond and be compatible with existing emergency operating procedures.

During the course of our discussions, several comments and suggestions were made which the staff may want to consider as these programs develop further. These are listed below. We do not request any formal response to this letter.

1. Some of the data used in NUREG-1150 (e.g., failure probabilities of motor-operated valves for certain postulated accidents) are now recognized to have been inappropriate. It appears that the results of NUREG-1150 are being used, at least to some extent, in evaluating the results reported in the IPEs and that some of the IPEs may be using the same inappropriate data used in NUREG-1150. It might be useful to reevaluate some of the originally reported NUREG-1150 results in light of new, more appropriate, data.

2. It also might be worthwhile to conduct sensitivity studies to determine the effects of using faulty data on IPE results.

3. Some of the IPEs will describe plants that have implemented the Station Blackout Rule (10 CFR 50.63) and some will describe plants that have not. It would be instructive if the results of the IPEs could produce an estimate of the risk

reduction achieved by implementing the rule. Even if the conclusion is that it is not possible to do this, that fact would be significant.

4. The NRC staff has made and is continuing to make detailed inspections of nuclear power plant maintenance programs. It would be useful to determine if there is any observable correlation between a good maintenance program and a low value of core damage frequency (CDF) as indicated by the IPEs. Dr. Murley's February 6, 1992 letter to the New York Power Authority regarding the FitzPatrick IPE seems to indicate that he feels there should be a detectable correlation between calculated CDF and "operability problems and procedural deficiencies."

5. It would be desirable to document and preserve the plant-specific PRAs that result from the IPE process. It was not clear whether the data base that is being compiled by Brookhaven National Laboratory will accomplish this. If it does not, other methods of doing so should be explored.

6. We suggest that, in addition to those things that have been identified by licensees, consideration be given to identifying, as vulnerabilities, any risk-significant sequence which has a large uncertainty in its upper bound.

7. Rather than treat shutdown risk on a generic basis, as appears to be the proposed approach, it should be treated on an individual plant basis, because it is likely to be very plant-specific. A mini-IPE might be appropriate.

8. It appears, on the basis of our discussion, that the staff has not yet arrived at an agreed-upon definition of an accident (in the context of accident management). Absent such a definition, there may be unproductive duplication between "accident management" and "emergency operating procedures." We recommend that an effort be made to arrive at a definition which will be acceptable to both the staff and to NUMARC.

Sincerely,

David A. Ward
Chairman

<u>References</u>:

1. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990
2. Letter dated February 6, 1992, from T. E. Murley, Office of Nuclear Reactor Regulation, NRC, to J. C. Brons, New York Power Authority, Subject: Request for a review of the FitzPatrick IPE with respect to the NRC's Diagnostic Evaluation Team Report

June 10, 1992

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

Dear Mr. Taylor:

SUBJECT: REGULATORY GUIDES 8.7, REVISION 1, AND 8N6 TO IMPLEMENT THE REVISED 10 CFR PART 20

During the 386th meeting of the Advisory Committee on Reactor Safeguards, June 4-5, 1992, we discussed the two subject regulatory guides related to the implementation of the revised 10 CFR Part 20, "Standards for Protection Against Radiation." Our Subcommittee on Occupational and Environmental Protection Systems and a Working Group of the Advisory Committee on Nuclear Waste (ACNW) discussed these guides, together with four other guides in this area for which the ACNW has the lead responsibility, during a joint meeting on May 27, 1992. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Our general comments on these two regulatory guides are summarized below.

1. Regulatory Guide 8.7, Revision 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"

   This guide contains detailed instructions on filling out NRC Form 4 (Lifetime Occupational Exposure History) and NRC Form 5 (Occupational Exposure Record for a Monitoring Period). The staff has done a creditable job of dealing with extensive public comments on the draft guide. We endorse publication of this guide for industry use.

2. Regulatory Guide 8N6, "Planned Special Exposures"

   This guide deals with infrequent, preplanned radiation exposures in excess of routine regulatory limits, deemed necessary because of some exceptional circumstances. The staff has done a creditable job of dealing with extensive public comments. Although we endorse issuing this guide for industry use, we suggest that you consider the desirability of holding a series of workshops to explain the intent of this guide to industry, to the unions that represent the relevant workers, and to the regional offices that will have to make judgments about adherence to the revised 10 CFR Part 20.

Sincerely,

David A. Ward
Chairman

References:

1. Memorandum dated April 10, 1992, from Bill M. Morris, Office of Nuclear Regulatory Research, NRC, for Raymond F. Fraley, ACRS, Subject: ACRS Review of Final Regulatory Guide 8.7, Revision 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," with enclosures.
2. Memorandum dated May 18, 1992, from Bill M. Morris, Office of Nuclear Regulatory Research, NRC, for Raymond F. Fraley, ACRS, Subject: ACRS/ACNW Review of Three Final Regulatory Guides to Implement Revised Part 20, with enclosures.