NUREG-1125
Volume 23

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2001 Annual

U. S. Nuclear Regulatory Commission

August 2002

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www nrc gov/reading-rm html.
Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices, inspection and investigative reports, licensee event reports, and Commission papers and their attachments

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1 The Superintendent of Documents
U.S. Government Printing Office
Mail Stop SSOP
Washington, DC 20402–0001
Internet bookstore gpo gov
Telephone· 202-512-1800
Fax: 202-512-2250
2. The National Technical Information Service
Springfield, VA 22161–0002
www.ntis gov
1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:
Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U S Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail DISTRIBUTION@nrc gov
Facsimile· 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www nrc gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—
American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi org
212–642–4900

Legally binding regulatory requirements are stated only in laws, NRC regulations; licenses, including technical specifications, or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2 206 of NRC's regulations (NUREG–0750).

NUREG-1125
Volume 23

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2001 Annual

U. S. Nuclear Regulatory Commission

August 2002

ABSTRACT

This compilation contains 54 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2001. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 4, is included by reference only. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

PREFACE

The enclosed reports, issued during calendar year 2001, contain the recommendations and comments of the U. S. NRC's Advisory Committee on Reactor Safeguards on various regulatory matters. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
13	Calendar Year 1991
14	Calendar Year 1992
15	Calendar Year 1993
16	Calendar Year 1994
17	Calendar Year 1995
18	Calendar Year 1996
19	Calendar Year 1997
20	Calendar Year 1998
21	Calendar Year 1999
22	Calendar Year 2000

<u>ACRS MEMBERSHIP (2001)</u>

CHAIRMAN:	Dr. George E. Apostolakis, Professor Massachusetts Institute of Technology
VICE CHAIRMAN:	Dr. Mario V. Bonaca, Retired Northeast Utilities
MEMBERS:	Dr. Thomas S. Kress, Retired Oak Ridge National Laboratory
	Mr. Graham M. Leitch, Retired PECO Energy, and Maine Yankee Atomic Power Company
	Dr. Dana A. Powers Sandia National Laboratories
	Mr. John D. Sieber, Retired Duquesne Light Company
	Dr. William J. Shack Argonne National Laboratory
	Dr. Robert E. Uhrig, Professor University of Tennessee
	Dr. Graham B. Wallis, Professor Dartmouth College

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

January 11, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Chairman Meserve:

SUBJECT: ISSUES ASSOCIATED WITH INDUSTRY-DEVELOPED THERMAL-HYDRAULIC CODES

We are responding to the Staff Requirements Memorandum dated November 7, 2000 [Reference 1], in which the Commission requested the ACRS to provide a more detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and to provide more specific recommendations on how these weaknesses should be addressed.

In the Background Section we explain why thermal-hydraulic codes are of high topical interest to the ACRS and the Commission. We then respond to the Commission's question about the effects of thermal-hydraulic codes on NRC's regulatory role. Finally, we list several specific recommendations which are cross-referenced to a discussion in the appendix. The bases for the recommendations are also included in the appendix along with specific items on slides cited by the Commission in its SRM [Reference 2].

Background

Thermal-hydraulic codes have been used for decades. Most of these codes have included deliberate conservatisms or bounding assumptions to address uncertainties in the models. The codes have proven to be adequate to satisfy regulatory requirements, when used with appropriate conservatism and judgment and when extensively examined by the staff. The efforts by the industry to improve the performance of its plants (by power uprates for example) and the efforts to reduce unnecessary burden pose new challenges to the use of these codes. It is no longer sufficient for a code to make conservative predictions; there is an increasing need for "realistic" predictions. In the case of new reactor designs, a lack of confidence in the proper application of first principles may require additional testing and benchmarking to establish regulatory confidence.

The replacement of conservative codes by realistic (best-estimate) codes raises a number of questions such as:

- How is the degree of realism to be measured?
- How do approximations, models, and assumptions affect the results?
- How are these approximations related to measures of realism such as bias and uncertainty?
- What other qualities, such as completeness and flexibility, are desirable in these codes?

When codes are relied upon to predict success criteria for risk analysis and to justify changes in regulations, further questions arise such as:

- Is the code giving the wrong answer and what is the impact of the result?
- If conservatism is reduced, what measure of confidence in code predictions is needed to ensure that adequate safety margins are preserved?
- Should more rigorous comparisons with data (assessments) be required? How much do these comparisons depend on the nature of the regulatory decision to be made?
- What is the effect of model uncertainty in the thermal-hydraulic codes on conclusions to be drawn from PRAs?

The Commission needs to be assured that the staff can respond to these sorts of questions and has the insight to pose others that may be important. The agency also needs to have a good perspective of the capabilities and limitations of codes and their modes of use so that it is on sure ground as it makes decisions.

We have recently been involved in reviewing realistic codes submitted by developers and these reviews are continuing. We have also reviewed a new Draft Regulatory Guide (DG-1096) [Reference 3] and Standard Review Plan Section [Reference 4]. We have learned some lessons and made some observations that were the background for our presentation to the Commission on October 6, 2000. At that time, we did not make specific recommendations. The recommendations that we now make are mostly for staff actions in response to the situation as we see it today. The final recommendation addresses the changing regulatory environment in a more general way, but is perhaps the most important of all.

Before presenting our recommendations, we address the question of how the perceived weaknesses of the thermal-hydraulic codes may affect the agency's regulatory role.

Effects of Codes on NRC's Regulatory Role

Traditionally, codes have been used to predict the behavior of plants under design basis and beyond-design-basis conditions. Because of the limitations of these codes, there is a probability that their predictions may be wrong; in other words, a better code would lead to a different answer to a safety question. The traditional approach has been to account for uncertainty through conservatism and additional safety margins, thus giving added assurance that the prediction of success is valid.

In contrast to the traditional approach, risk-informed regulation is based on realistic analysis and seeks to avoid undue conservatism. Codes are used to evaluate success criteria for risk analysis. With this approach, the uncertainties must be explicitly taken into account to provide a desired level of confidence in the results. Similarly, power uprates will depend on the ability of more realistic codes to justify the reduction in conservatisms required to provide the necessary margin.

Because the theoretical basis of two-phase flow thermal-hydraulic codes is incomplete, significant uncertainties in predictions are expected. They can be reduced by developing better models, and better quantified by more extensive experiments. If the theoretical basis is weak, more experiments are needed to support empirical correlations and to establish the uncertainties.

The key question is: "What quality of codes is needed to support regulatory decisions?" The rational measures of "quality" are the uncertainties in the code predictions and the influence these uncertainties have on decisions.

Our concern is that codes may not have sufficient quality to support decisions that the Commission has to make. More specifically, poor quality codes may restrict the degree to which the regulations may confidently be risk informed and the extent to which conservatism may be reduced.

The use of thermal-hydraulic codes affects the performance goals of the agency in the following ways:

- Maintain Safety

  If uncertainties in code predictions are not adequately understood and addressed, safety may be (unwittingly) compromised.

- Increase Public Confidence

  Excessive uncertainties, errors, and unjustified assumptions in codes reduce public confidence, and more importantly the confidence of the informed technical community, including the NRC staff and the users of the code.

- Increase Efficiency and Effectiveness

  Weaknesses in code documentation and validation lead to lengthy negotiations between the staff and the applicant and substantially increase the time and effort for decisionmaking

  Poor theory leads to requirements for more extensive experimental evidence.

- Reduce Unnecessary Burden

  Large uncertainties in code predictions lead to conservative decisionmaking in order to maintain assurance of safety. This imposes burdens that better predictions might show to be unnecessary.

Recommendations

(The item numbers in the appendix to which these recommendations respond are given in parentheses.)

To ensure that the codes can meet present and anticipated standards of quality:

1. The staff should make clear that standards to be applied to documentation of proprietary codes are the same as for codes generally accessible to public scrutiny (Items 2, 10, 11, 14).

2. The staff must continue to require that vendors and licensees supply working versions of the codes for internal NRC use and evaluation (Items 6, 7, 12, 18).

3. The staff should recommend how improvements can be more readily incorporated into codes (Items 1, 20, 21).

To promote more effective evaluation of the codes:

4. The staff should continue developing its own thermal-hydraulic code, making it more reliable, flexible, and easy to use (Items 1, 9, 17, 19).

5. The staff should examine ways in which the process of evaluation and assessment of proprietary codes can be made more publicly accessible and scrutable (Items 2, 11, 12, 14, 19).

6. The staff, perhaps in cooperation with an industry-supported entity such as the Nuclear Energy Institute (NEI), should undertake an authoritative study assessing when, how, and why codes produce reasonable results despite numerous assumptions and simplifications. This study should include measures of code strengths and weaknesses and include an assessment of circumstances under which the shortcomings of the codes may have significant influence on regulatory outcomes (Items 1, 3, 4, 5, 19).

7. The staff should take steps to ensure that the existing data base for thermal-hydraulic code evaluation is preserved in accessible form (Items 4, 8, 12, 16).

To ensure that the codes meet anticipated regulatory requirements:

8. The staff should consider how definite measures of code quality, such as bias and uncertainty in predicting significant phenomena and success criteria, can be more specifically required as outputs from the code assessment process (Items 8, 12).

9. The staff should investigate and recommend how uncertainties in code predictions can be best quantified to be suitable for incorporation into risk-informed regulation (Items 8, 12, 13, 15).

10. The staff should reevaluate the design specifications for the outputs of codes and their relationship to present and anticipated regulatory requirements (Item 6, 7, 8, 13, 22).

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated November 7, 2000, from A. L. Vietti-Cook, Secretary, to J. T. Larkins, ACRS, Subject: Staff Requirements, Meeting with Advisory Committee on Reactor Safeguards, October 6, 2000, Commissioners' Conference Room, One White Flint North, Rockville, Maryland, (M001006B).
2. Slide presentation during ACRS Meeting with NRC Commissioners on October 6, 2000, "More Realistic Thermal-Hydraulic Codes."
3. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide, DG-1096, "Transient and Accident Analysis Methods," dated July 18, 2000.
4. U.S. Nuclear Regulatory Commission, Standard Review Plan Section 15.0.1 (subsequently changed to 15.0.2), "Review of Analytical Computer Codes," dated April 14, 2000.

Attachment: Appendix

Appendix

Discussion of the specific points raised in the ACRS slides addressing issues associated with industry-developed thermal-hydraulic codes is provided below.

1. **Many codes have the same ancestry, including a 30-year old foundation**

Several codes are derivatives of the RELAP series, going back to developments at Aerojet Nuclear Corporation (ANC, now INEEL) in the late 1960s, with influence from the Bettis FLASH code. Several assumptions were made at a fundamental level in order to produce equations that could be solved with the computer technology of the time. These assumptions introduced errors and uncertainties that have not been quantified for general purposes, though for some applications the results appear to be adequate. For example, the staff responded [Reference 5] in 1972 to a "controversy concerning the momentum equation" by saying:" We find that the relatively few assumptions made by ANC (in the RELAP3 code) can all be justified in LOCA analysis. There may be further simplifying assumptions which could be justified. However, there is no adequate basis now available to us for judging the adequacy of the assumptions implicit in the momentum flux representations used in the four vendor evaluation models."

In the Water Reactor Evaluation Model report [Reference 6], the staff lists several options of the momentum equation, each of which is unrealistic to some degree, as a basis for modeling real reactor components. Dr. Novak Zuber, an ACRS consultant who was an AEC staff member at the time, informs us [Reference 7] that in 1974 he reviewed RELAP4 and concluded that its momentum equation was incorrect. It was not corrected at the time because the effect on peak clad temperature for a large-break LOCA appeared to be small. Emphasis in the subsequent decades seems to have been on getting codes to run and introducing correlations to describe the details of the phenomena without reviewing the basic equations and their limitations.

Some of the original controversial, or at least approximate, equations are still in use, sometimes expanded or rederived by unconvincing methods. Some of the examples used to illustrate them appear to be incorrect, decreasing the confidence of a reviewer in the approach used.

The point here is that there is a long history of decisions having been made to accept what was perhaps the best available or "adequate" model, despite what appear to be fundamental errors in the basic formulation; the consequences, as the uses of codes expanded, were not well understood.

One consequence of this common ancestry is that if the NRC code has the same basic assumptions as a vendor's code, and no attempt is made to investigate alternatives, then a dimension of "independence" may be sacrificed.

2. **Designed specifically for nuclear applications. Not commercial or academic.**

Computer methods have evolved rapidly in recent decades in all branches of science and engineering. An example is computational fluid dynamics (CFD) codes that are now widely used by industry and universities. Feedback from users and continual upgrading by code vendors spurs development, corrects faults or limitations, and improves capabilities and accuracy. Since the basis of these codes is transparent, or is available to the more sophisticated user, errors and shortcomings are discovered and improvements are made at many levels of detail.

In contrast, nuclear codes, particularly those that are treated as proprietary, have not been subject to such public scrutiny. The industry has been reluctant to improve the codes, even when this is desirable, perhaps because of the requirements for regulatory review and the cost of licensing changes.

3. **Contain many assumptions, idealizations, "best-shot" estimates, and user choices.**

Code documentation is usually presented so that the derivations appear to follow a logical thread based on sound fundamentals. Despite increases in computing power, it is still necessary to introduce assumptions. Sometimes assumptions are explained, but often they are not justified by further rationale. Other assumptions may be inherent in the derivations but are not acknowledged and are only apparent if a reviewer makes the effort to understand what is being done. It may then be discovered that the derivations have a much less authoritative base than is implied by the text.

For example, most codes make use of a flow regime map defining how the two-phase flow pattern changes from "bubbly" to "annular" (a liquid film on the wall) to dispersed droplet regimes as the fraction of vapor increases.

These flow regimes are dependent on the geometry of the devices through which the fluids flow. For instance, they are different for bends and inclined pipes. They are also considerably influenced by conditions upstream and downstream of the component under consideration. For example, if fluid is introduced into a pipe along the wall, it may take a considerable distance before it is entrained into an established droplet pattern.

Even the many flow regime maps available for two-phase flows in long straight pipes are often not compatible with each other, and data can always be found that disagree with the best versions.

There is little record that flow regimes have actually been measured and the results compared with predictions for reactor system components such as rod bundles, downcomers, or the upper and lower plenum of a reactor vessel. The sensitivity of predictions to the choice of flow regime boundaries is usually not investigated. The choice of a simple universal flow regime map as the basis for a code represents a huge simplification that can only be justified by arguments that it "works" well enough for the purpose of predicting some overall parameter of interest under some specific circumstances. Under other circumstances, such as a design change or a new concern (such as boron dilution) in which other regimes could play a significant role, the code must again be shown to "work" or be modified.

Even after a flow regime is selected, additional simplifications are necessary. Often these simplifications are then applied in a questionable manner. Derivations may be made for "one-dimensional flow" in a straight pipe and then applied to bends or other reactor components in ways that are not explained or justified. Correlations of wall friction factors or forces between the vapor and liquid that were developed as very rough approximations for air-water flows in straight pipes are adopted to represent steam-water flows at high pressure in a host of different geometries. Sometimes the user is given a choice of correlations without clear guidance about which one to select, and this introduces a "user effect" whereby different answers can be obtained to the same problem.

When no precedent exists for modeling a component or situation, the code developer may make an estimate based on an idealized view of what is happening. Without independent experimental validation of the individual model, the reviewer cannot tell how reasonable the estimate is except in the context of the code's apparent success at a global level, which could be the result of compensating errors.

We do not mean to imply a need for excessive detail, if this can be shown to be inappropriate. Professional judgment must guide choices among imperfect, but adequate, analytical models. Engineers who need to get results with limited knowledge, resources, and time must often make such choices. The point is that it is unwise to rely on the resulting structure without a broad base of methodical assessment against the full range of conditions for which the code will be applied. It also means that restrictions must usually be placed on the use of a code in unevaluated situations, or for new or modified designs for which a sufficient basis for assessment does not exist, or to answer questions for which greater accuracy is required than has been demonstrated. These characteristics of codes demand detailed staff review, supported by knowledge, persistence and sufficient resources.

4. Codes have evolved, but the development process is hard to trace.

The codes have been developed over the years by various authors, many of whom are no longer active. The rationale for the features and assumptions in the codes, their advantages and limitations, and why a certain approach was chosen may have been forgotten. When the rationale is lost, the present generation of code developers, users, and reviewers has to rediscover and reevaluate the technical bases of these features. More detailed explanations and justifications in the original documentation might avoid the need for this extra work. This situation supports the need for the staff to have its own code and to maintain a clear record of why design choices were made in its development.

5. Codes may "work" but they are not based on a mature, secure science.

We discussed some of the rationale for this statement under Item 3. The science of multiphase flow and heat transfer has not reached a point where predictions can be made solely from a basis of secure fundamentals (as they can for many viscous single-phase flows, for example). Codes have evolved as an elaborate tapestry of interwoven working assumptions and approximate equations and correlations that have proved to be useful. Longevity of these engineering methods is no assurance of maturity, nor does it guarantee that the codes need no further development and improvement as new questions arise. Nevertheless, a series of demonstrations of applicability to reactor transients has established confidence in their utility as one input to regulatory decisionmaking, as long as their limitations are adequately understood and evaluated.

6, 7. Pro: For several accidents (e.g., small-break LOCA) a few phenomena appear to dominate and the calculated output figures of merit, such as peak clad temperature, appear insensitive to the details. Con: Some phenomena that could be important are poorly modeled (e.g. two-phase level in core boil-off).

We present the following examples to make the point that the precision of code predictions, and their sensitivity to details, depend on the question being asked. After the water level drops during the initial transient, small-break LOCA becomes "a pot of boiling water with steam escaping from a small hole." A relatively simple model may be quite successful in representing the main phenomena. The steam release rate is given by an energy balance for the vessel and the system pressure is governed by single-phase steam flow out of the break. On the other hand, the swelling of the two-phase mixture in the core and the upper plenum is a complex phenomenon. In the analysis of one experiment, for example, it could only be represented accurately by changing the interphase friction factor by a factor of five from the value typically used in the code chosen for the analysis. Such discrepancies are not unusual when dealing with two-phase phenomena. The net effect on peak clad temperature was smaller than this discrepancy suggests, but the result could have a considerable impact if the success criterion was that the core be completely covered by the boiling pool. We note that boilup and steam cooling are considerations in licensee proposals for changes to the current licensing basis. For example, boron mixing and the recent proposal to use low pressure systems as an alternative strategy in response to a fire. These examples point out the importance of the staff having the wisdom and experience to know when to ask for more information and when to accept a calculation that is offered. This decision may involve input from the research branch of the agency.

8. **Code predictions have to be assessed for each application and extensive sensitivity checks performed.**

Because of the uncertainties already mentioned, a code cannot be approved *a priori* for all applications. The staff must be satisfied that the range of data for which the code "works" covers the range that is needed to provide adequate confidence that the results can be extrapolated to predict the performance of a full-scale system during the relevant scenarios. If conservatism is to be reduced, better confidence in the code is needed, in the form of reduced uncertainty in its predictions. One of the ways to assess this uncertainty is by assigning distributions to the assumptions and parameters that go into the code and propagating these through the calculation of the desired result. Another way is through quantitative evaluation of the accuracy and uncertainties in the code by systematic statistical comparisons with data. We perceive a need for the staff to be more specific about what are acceptable methods of deriving and expressing the uncertainties in codes and how these methods are to be used in the regulatory context.

9. **The staff's knowledge, experience, and thoroughness are key.**

For all of the above reasons, decisions about the acceptability of a code and its range of validity depend heavily on the competence and thoroughness of the staff. Steps are currently being taken, through the preparation of a Regulatory Guide (DG-1096) and Standard Review Plan Section (15.0.2) to develop more explicit explanations of the expectations of the staff and criteria to be used for evaluating codes. This should reduce variability in the extent and depth of reviews. It should also help submitters to understand the standards to be used. We have commented on draft versions of these documents.

The Commission needs to be assured that the staff is sufficiently consistent and thorough in its reviews and has sufficient resources and capabilities to complete them.

10, 11. **Some code documentation is poor. The physical basis for analytical models is often incomplete and poorly explained.**

The above, more general observations that we cited under Item 3, did not prepare the ACRS for what it found in some of the code documentation. Examples are given below. The specific submittals in which these shortcomings were observed will not be identified, but no code was without some of them.

- Basic equations, of the type to be found in standard textbooks, contained many major typographical errors. These should be immediately apparent to a knowledgeable observer.
- Equations were so garbled as to require lengthy rederivation by the reviewer.
- Simple algebraic errors appeared to be made in deducing one equation from another. The result was an equation which could not have been derived from the given starting point.
- Scalar and vector quantities appeared to be intermixed in inappropriate ways.
- Examples intended to illustrate the method being used actually served to discredit it.
- Terms in equations were insufficiently defined. Examples of the use of these equations suggested a lack of understanding of how the terms should be evaluated.
- General derivations were made that appeared incompatible with specific cases. Such derivations were usually unique to the particular code.
- The logic in developing coefficients or expressions for use in the code was unclear. Equations were written down without an explanation of how to use them or where the terms come from. A reviewer could not tell what process was actually being followed.
- Assumptions were made on an *ad hoc* basis without supporting evidence.

- Equations were derived for straight pipes with no indication of how they might apply to the real shapes found in actual systems.

Shortcomings of this sort would be much less likely if the codes were prepared and reviewed for open publication. The ACRS would prefer to see the same standards applied and the same quality achieved whether or not the code will be publicly available.

Despite these deficiencies, the codes have proven to be acceptable to satisfy regulatory requirements. However, an important aspect of the review process is the establishment of confidence in the approaches taken by the code developer. Weak documentation undermines this confidence.

12. Assessment is unfocussed and insufficiently extensive.

Assessment is the comparison of predictions with data to determine how well the code works. Because of the many assumptions and approximations made in the codes and the numerous empirical coefficients and correlations imported from other applications, it is important that comparisons be made with data from actual or scaled nuclear components. Some codes are presented for review with few, if any, such assessments.

Other code predictions are compared with a small selection of data. For instance, predictions are compared with some results from a Loss of Fluid Test (LOFT) facility test. It is not explained why this test was chosen and why predictions were compared with only a small group of data when other results were also available.

Code assessment should be a more complete and logical process. Sensitivity studies can reveal which parts of the code it is most important to evaluate. Comparisons can then be focused on those parts and a thorough evaluation made, covering the whole range of parameters expected in practice, with an emphasis on discovering limitations, rather than showing some limited general agreement. A logical road map should be provided, explaining what features of the code are to be assessed by a particular comparison and how they are being evaluated. For the purposes of risk-informed regulation, code assessment should result in quantitative measures of uncertainty so that the risk that the code will give an unacceptable answer can be determined.

Code assessment is presently one of the weakest links in the NRC's review process. Insufficient thought has been given to demonstrate acceptability, leaving too much up to the judgment of the NRC reviewer.

We believe that a logical assessment review process can be developed. It should start with identification of which features require assessment and to what degree. This might resemble a Phenomena Identification and Ranking Table (PIRT) process, but, unlike PIRT, it should be carried through to the end of the assessment review to determine not only what a code should do but how well it does it. A procedural framework should be set up explaining how specific features are to be assessed and by what measures. At present, in some assessments, there is only a tenuous association between the selected comparisons with data and specific questions to be answered.

The assessment process described in the Draft Regulatory Guide (DG-1096) is qualitative and leaves considerable latitude for interpreting how to measure success. This essential process would be considerably strengthened, without being unnecessarily prescriptive, if specific quantitative outputs were identified. For the purpose of regulation, including risk-informed considerations, the most significant of these are measures of bias and uncertainty in predicting significant phenomena and success criteria.

13. **Methods for calculating uncertainties are primitive and not comprehensive.**

The regulations require an assessment of uncertainty for all "realistic" codes. We have encountered a wide range of approaches to estimating uncertainties. At the crudest level, a predicted curve is shown to pass in the vicinity of some data points and agreement is characterized by terms such as "good" or "acceptable." This is not a basis for quantitative assessment of model uncertainty and its effect on safety margins.

Another approach is to vary key parameters in the code and determine the effect on the success criteria, such as peak clad temperature. This gives useful information about sensitivities of the code to these specific parameters. It does not capture additional uncertainties due to the overall structure of the code and the particular form of equations and correlations chosen. Moreover, without comparison with data, the actual likelihood that parameters will have specific values within the chosen range is unknown.

The most thorough approach we have seen to date methodically compares predictions with many data points to quantify bias and uncertainty in predicting specific phenomena. The uncertainties are then combined to quantify the likelihood of meeting success criteria in the regulations. This impressive achievement follows the intent of the Code Scaling, Applicability, and Uncertainty (CSAU) Evaluation methodology [Reference 8]. However, questions can be raised about the completeness of the implementation. For example, errors in void fraction, traced to uncertain forces between the steam and water phases, may be evaluated from data taken predominately in a large vertical duct or vessel. How does this apply to flows in more tortuous passages, as in the reactor core, in horizontal flows, or flows in bends such as the loop seal in a PWR primary circuit?

Because of the large variability in approaches to the evaluation of uncertainty, and the deficiencies in present methods, the staff should conduct a comprehensive study of ways to quantify and assess uncertainty, the measures to be employed, and their adequacy for meeting regulations. This will require innovative research to show how to evaluate uncertainties when formulating important equations and correlations so that the uncertainties are incorporated into the solution routine of the code itself.

In discussing the uncertainty methodology, draft Regulatory Guide DG-1096 refers to a previous Regulatory Guide, 1.157 [Reference 9]. Regulatory Guide 1.157 covers many features of realistic calculations, including the evaluation of uncertainty and bias, but its guidance is very qualitative and leaves considerable latitude in interpretation. Typical phrases seen are "performs adequately," "acceptable provided their technical basis is demonstrated with appropriate data and analysis," "uncertainties and biases in the parameter should be stated," and "sensitivity studies should be performed." The meaning of these requirements depends on how the code will be used in making regulatory decisions. Without a clear connection between the outputs from the code and their regulatory function, there is no basis for deciding when the modeling techniques are good enough. This is why we believe that the staff needs to clearly tie its safety evaluation findings to the specifications these codes need to satisfy, particularly to support risk-informed regulations.

14. Documentation should be acceptable to knowledgeable, impartial observers.

When a code is proprietary, the vendor and the NRC staff are usually the only people who have access to its documentation. The ACRS may be the one external group that provides a third-party review; it becomes the sole independent guarantor of public confidence. At the same time, the ACRS does not have the resources to investigate all the details in these codes.

As already described, we were surprised to find that some documentation had not been prepared to standards appropriate to inspiring confidence in impartial observers. There are various reasons for this situation, some historical and traceable to the way the regulatory process has functioned. Eventually, perhaps through published papers based on the work, or through a relaxation of proprietary sensitivity, the documentation will be seen by knowledgeable outsiders. Moreover, it is professionally demoralizing to both industrial engineers and to the NRC staff to discover uncorrected errors, some of which are obvious from a cursory review, in approved documentation. The staff should make clear that standards to be applied to documentation of proprietary codes are the same as for codes generally accessible to public scrutiny.

15. Risk-informed regulation will require more quantitative evaluation of model uncertainties and their consequences.

Shortcomings in the codes may have been allowable in the past when "engineering judgment" imposed appropriate conservatism and safety margins to compensate for lack of confidence in accurate predictions.

The basis of risk-informed regulation is different. Predictions are "realistic" but not exact. The probability of errors in code predictions must be assessed. Safety margins require a more precise definition so that the probability of exceeding them can be evaluated.

Just as in the case of PRAs, the quality of thermal-hydraulic codes must be compatible with the purposes for which they will be used to make decisions. The need for increased code quality measured by decreased uncertainty will grow, because requests for reduction in regulatory burden, power uprates, digital instrumentation and control, and the synergistic effect of such changes on aging plants may erode margins of safety.

16. The data base for assessment must be preserved and, in some cases, expanded.

The decisions that went into formulating deterministic regulations, such as Appendix K to 10 CFR Part 50, were confirmed to a great extent by the experience with specially designed experimental facilities such as LOFT.

The Westinghouse AP600 passive plant design differed substantially from previous designs. Confidence in its characteristics could not be obtained from theory alone but required tests in several facilities, notably the scaled APEX facility at Oregon State University (OSU).

As new questions arise or old questions become more critical, the data base is the arbiter. When the data base is inadequate new experiments are conducted, as in the thermal-hydraulic test program related to pressurized thermal shock being performed by the NRC at OSU.

We are concerned at reports that parts of the thermal-hydraulic data base, developed at considerable expense, that might be sufficient to answer questions raised by suggested changes in regulations or plant characteristics, are unavailable or have been lost for reasons such as changes in methods of computer storage. Steps should be taken to inventory and maintain this data base and ensure its preservation in usable form.

17. A base of experts needs to be maintained.

We consider three groups of experts to be important: the code developer, the NRC staff, and the experienced public professionals who may be called upon for independent review, consultation, or participation in expert panels.

Our experience with reviewing codes suggests that a very small group of developers is sufficiently knowledgeable to answer technical questions. Some of the original developers have left the industry. Perhaps the notion that codes were mature and needed little more attention has taken resources away to the point where some vendors may be insufficiently prepared for the effort required to upgrade codes to the "realistic" level. The NRC might consider conducting periodic audits of industrial support for codes of the type that were performed by the agency in the past. These audits focused on such important issues as code quality assurance and the adequacy of documentation.

The NRC needs two sets of experts on its staff. The first group reviews developers codes and runs them with sufficient insight and informed curiosity to make regulatory decisions. The second group, in the research category, makes independent assessments with NRC's own code, investigates "what if" scenarios, anticipates future uses that may stretch the capabilities of existing tools, and develops new capabilities when these are required. Our impression is that these teams are presently close to minimal strength, which may delay efficient processing of applications involving new codes.

The third group of experts has been well served by those who were active in developing thermal-hydraulic theory and codes in the 1960s and 1970s. Many of these people are now retired and may soon no longer be available. The NRC should encourage the development of a new generation of experts in this field.

18. Staff should run and evaluate vendor codes independently.

The former process by which the NRC staff evaluated code predictions submitted by the developer, had many limitations. Obtaining results other than those selected by the developer, exploring the effect of different assumptions, determining limitations on the range of variables or conditions for which the code performed well, and getting other information needed for a thorough assessment, was laborious and sometimes frustrating.

In the past two years the staff, with ACRS encouragement, has adopted a policy of obtaining working versions of these codes and running them. This is a much more efficient way of discovering the strengths and robustness of a code, as well as its limitations and weaknesses. It also allows the staff to check what could not be checked previously, that the code, as programmed, actually follows the documentation. This sort of openness vastly increases the effectiveness of the review process and should benefit both parties, as well as enhance public confidence.

19. **Staff should maintain in-house code competence, including an NRC-developed code.**

There is no standard code for thermal hydraulics. Each code has its individual characteristics, even peculiarities.

The ACRS strongly supports the current efforts by the NRC research staff to combine its present suite of thermal-hydraulic codes into one, and to improve the models, functions, flexibility and speed of operation. For the foreseeable future, there will exist manifold uncertainties about the effects of models in vendor codes, their limitations, phenomena that may be insufficiently explored and undiscovered implications. The NRC needs its own tool it is familiar with, can experiment with, and use to anticipate new questions and resolve them. The staff will then be able to accept or question features of the developers codes based on independent knowledge. It will also have a tool readily available for assessing new concerns, major plant events, and operating transients.

The informed technical community has long been aware of shortcomings in thermal-hydraulic codes, even though these codes have proven adequate to satisfy current regulatory requirements. A prime example is the various approximations and shortcuts that are needed to manipulate the momentum conservation equation into usable form.

With access to vendor codes and intimate knowledge of its own code, the staff is in the best position to obtain a reasoned perspective on this issue. With the enormous advances in speed of computation, including parallel processing and improvements in code architecture, it should be possible to run a set of carefully designed computer experiments to assess the practicality and limitations of existing methods. For example, the consequences of using alternative forms of the momentum equation, doubling or halving approximate terms, and including or neglecting various effects could be realistically assessed. The aim would be to produce an authoritative document that would enhance public confidence and provide a landmark source of reference in future decades. This could be a major contribution to the effectiveness and efficiency of the NRC review process and the preparation of documentation by industry.

A possible impediment to this sort of comprehensive evaluation is the industry wish to protect its proprietary codes. This may inhibit the public availability of independent assessments of technical details. Perhaps this impediment could be overcome by a cooperative effort between the staff and an industry-supported entity, such as NEI. There are recent precedents for this sort of cooperation.

20. Regulatory processes should encourage code improvements.

Several elements of codes are based on ideas that originated 30 or even 40 years ago. They were tentative at the time and it is remarkable that they have not been replaced or improved. We are not sure why this is. There are mechanisms for modifying codes, but industry must usually sense some benefit before doing so. Perhaps the regulatory process itself has an inhibiting effect on modifications, once approval has been obtained. We recommend that the staff suggest effective ways to encourage the development of code improvements. The measure of success would be a reduction in uncertainty in the output of the code and in the resulting conservatism in decisionmaking.

21. NRC should be preparing for an eventual new generation of thermal-hydraulic codes.

Computer methods now permeate society and are one of the most rapidly evolving technologies. Computational fluid mechanics, which was the subject of research a couple of decades ago, is now a flourishing industry. Continually evolving commercial codes are commonplace in the arsenal of industrial engineers. Grids involving over a million nodes are routinely used to evaluate air flows over aircraft and automobiles or over the internals in a vehicle engine compartment. The NRC has a small research effort to use these codes for single-phase flow problems relevant to safety.

Comparable multiphase industrial thermal-hydraulic codes have also been under development but they are not as mature or reliable as their single-phase parents. However, commercial needs are stimulating a rapid evolution. The NRC clearly needs to keep abreast of these developments and decide which commercial codes, or which features of them, might profitably be incorporated into nuclear safety assessment in the future. The NRC was a leader in code development two decades ago; it now needs to adjust to a technological climate in which the most significant developments originate outside the agency.

The rewards from use of a new generation of commercial codes would be greater confidence in the results and a more efficient review process. If a code is widely used, it should limit the necessity for each licensee to subject the code to broad-based assessment for each application. The NRC might eventually be able to give generic approval to codes, reducing the time needed to examine the details of every new application.

22. Specifications for codes

Finally, we explain Recommendation 10, that the design specifications for codes be reexamined. We cite two examples of future anticipated needs. The examples are not intended to be exclusive.

Past codes were used to evaluate design basis accidents, particularly LOCAs, as described in 10 CFR 50.46 (the ECCS Rule). If the ECCS Rule were to be risk-informed, what would be the appropriate output from codes in order to support this new regulation? What would be the functional requirements of these codes in terms of quantitative measures of accuracy and uncertainty that are now left vague? What measures would be required to demonstrate that the codes were "realistic"? How would model uncertainties in the codes be incorporated into risk estimates in order to make them more realistic?

The Commission is receiving requests for power uprates on the order of 20%. This significant increase in core power must have some impact on safety. Can present codes quantify this impact? What maximum power uprate is tolerable and on the basis of what criteria? What must codes be able to do in the future to provide realistic measures of safety margins rather than the conservative prescriptive criteria used in the past?

We foresee that the resolution of these sorts of questions will be held up by inconclusive arguments about code quality and credibility unless the agency clearly defines what the specifications for codes must be to support anticipated regulations. This should be a creative and ongoing activity, involving both the research and the regulatory offices of the agency.

References:

5. Supplementary testimony of the AEC Regulatory Staff, Public Rulemaking on Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Power Reactors, Docket No. RM 50-1, dated October 26, 1972.

6. U.S. Nuclear Regulatory Commission, WREM: Water Reactor Evaluation Model, NUREG-75/056, May 1975.

7. Memorandum from N. Zuber, ACRS Consultant, to G.B. Wallis, ACRS, Subject: The Effect of Deregulation on NRC's Capabilities in the Field of Thermo-Hydraulics, dated April 6, 2000.

8. U.S. Nuclear Regulatory Commission Report, "Quantifying Reactor Safety Margins - An Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident", NUREG/CR-5249, December 1989.

9. Regulatory Guide 1.157. "Best-Estimate Calculations of Emergency Core Cooling System Performance", May 1989.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 1, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Dr. Travers:

Subject: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY

In a memorandum dated July 20, 2000, you requested that the ACRS examine the technical issues associated with a differing professional opinion (DPO) on steam generator tube integrity. Specifically, you requested that the ACRS function as the equivalent of an Ad Hoc panel, under Management Directive 10.159, to review the DPO issues and provide you with a summary report documenting the conclusions and any recommendations relative to the pertinent technical issues.

In a memorandum dated September 11, 2000, we informed you of the establishment of an Ad Hoc Subcommittee that would function under the provisions of the Federal Advisory Committee Act, to review the technical merits of the DPO issues and develop proposed positions for consideration by the full committee. The Ad Hoc Subcommittee was composed of ACRS members D. A. Powers (Chairman), M. V. Bonaca, J. D. Sieber and T. S. Kress, and R. G. Ballinger from the Massachusetts Institute of Technology. The Subcommittee was supported by three consultants hired by the staff: I. Catton, University of California at Los Angeles, J. C. Higgins, Brookhaven National Laboratory, and R. E. Ricker, National Institute of Standards and Technology.

During its meeting on October 10–14, 2000, the Ad Hoc Subcommittee met with J. Hopenfeld, the DPO author, Professional Engineer R. A. Spence of the staff, and several other members of the NRC staff to discuss the DPO author's contentions and the staff's responses. To support its review, the Subcommittee and its consultants reviewed a large volume of documents, including those referenced in the Subcommittee's report.

Based on its discussion of the DPO author's contentions and the associated NRC staff's responses, the Subcommittee developed a NUREG report documenting its conclusions and recommendations, along with the bases for arriving at these conclusions and recommendations. This report was reviewed by the members of ACRS and the consultants to the Ad Hoc Subcommittee.

During the 478th meeting, December 6-9, 2000, the ACRS reviewed the conclusions and recommendations of the Ad Hoc Subcommittee. The ACRS also discussed the DPO issues during its 477th meeting, November 2-4, 2000. During this review, the Committee had the benefit of discussions with the DPO author and representatives of the NRC staff. The ACRS also had the benefit of the documents referenced in the attached report of the Ad Hoc Subcommittee.

The ACRS endorses the Ad Hoc Subcommittee's conclusions and recommendations included in its report, which is being sent to you with this letter for use in resolving the DPO issues. The ACRS would like to be kept informed of the resolution of the DPO issues.

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

Sincerely,

Dana A. Powers

D. A. Powers
Chairman, Ad Hoc Subcommittee

*Enclosure: NUREG - XXX, "Voltage-Based Alternative Repair Criteria," a Report to the Advisory Committee on Reactor Safeguards by the Ad Hoc Subcommittee on Differing Professional Opinion, February 2001.

*See NUREG-1740 for the published version of this enclosure.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 8, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-152, "DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS"

During the 479th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2001, we reviewed the proposed resolution of Generic Safety Issue (GSI-152), "Design Basis for Valves that Might be Subjected to Significant Blowdown Loads." During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Conclusion

We agree with the staff's proposed resolution of GSI-152.

Discussion

In our letter dated November 20, 1989, we raised a concern that although a valve might meet the NRC-approved design bases, these design bases might not address the need for the valve to close against the differential pressure resulting from a large high-energy pipe break. To address this concern, the staff established GSI-152. Our concern was broader than that stated in GSI-87, "Failure of HPCI Steam Line Break Without Isolation" and Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." GL 89-10 specifically focused on the ability of motor-operated valves (MOVs) to operate under design basis conditions.

The staff issued Supplement 3 to GL 89-10 to provide guidance to licensees for ensuring the capability of containment isolation valves in the reactor water cleanup, high pressure coolant injection, and the reactor core isolation cooling systems in boiling water reactor plants to isolate the largest credible downstream pipe break.

The industry established programs to understand and correct valve operating weaknesses. Guidelines were issued for determining the design basis differential pressure for MOVs within the scope of GL 89-10.

In the early 1990s, the NRC staff conducted inspections of the licensees' GL 89-10 programs including evaluations of the design bases. These inspections confirmed that deficiencies in the design bases and in valve operating performance had been identified and corrected. These results were shared with the industry in Information Notices 96-48, "Motor-Operated Valve Performance Issues," and 97-07, "Problems Identified During Generic Letter 89-10 Closeout Inspections."

Based on the issuance of Supplement 3 to GL 89-10 and subsequent staff and industry initiatives, we support the staff's proposed resolution of GSI-152.

Sincerely,

George E. Apostolakis
Chairman

References :

1. Memorandum dated January 18, 2001, from Michael E. Mayfield, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Transmittal of GSI-152 Close-out Report.

2. Letter dated April 23, 1993, from Paul Shewmon, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Prioritization of Generic Safety Issue 152, "Design Basis for Valves that Might be Subjected to Significant Blowdown Loads."

3. Report to Kenneth M. Carr, Chairman, U.S. NRC from Forrest J. Remick, Chairman ACRS, Subject: Proposed Resolution of Generic Safety Issue 87, "HPCI Steam Line Break Without Isolation," dated November 20, 1989

4. U. S. Nuclear Regulatory Commission, Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989.

5. U. S. Nuclear Regulatory Commission, Generic Letter 89-10, Supplement 3, "Consideration of the Results of NRC-sponsored Tests of Motor-Operated Valves," dated October 25, 1990.

6. U. S. Nuclear Regulatory Commission, Information Notice 96-48, "Motor-Operated Valve Performance Issues," dated August 21, 1996.

7. U. S. Nuclear Regulatory Commission, Information Notice 97-07, "Problems Identified During Generic Letter 89-10 Closeout Inspections," dated March 6, 1997.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 9, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DRAFT ANS EXTERNAL EVENTS PRA METHODOLOGY STANDARD

During the 479th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2001, we met with representatives of the American Nuclear Society (ANS) External Events Working Group to discuss draft BSR/ANS-58.21, "External Events PRA Methodology Standard." We also had the benefit of the documents referenced.

The traditionally called "external" events, e.g., earthquakes, high winds, and external floods, have been found to be among the major contributors to risk for many plants due to the potential for dependent failures of plant safety systems. The assessment of these risk contributors requires the integration of a number of diverse technical disciplines and provision for the utilization of expert opinion. Because the occurrence of external events of sufficient magnitude to cause plant damage is rare and statistical evidence is sparse, expert judgment is required to develop the necessary probability distributions for risk assessments. The resulting assessments thus involve large uncertainties.

The ANS Standard does a good job in defining the requirements for a state-of-the-art assessment of the risk (including uncertainties) from external events. The commentary, including notes and references, that accompanies each requirement provides valuable information and guidance for meeting individual elements of the Standard. Thus, the ANS Standard resembles a traditional "design-to" standard.

An important feature of the proposed ANS Standard is that it was designed to be consistent with the standard for internal events under development by the American Society of Mechanical Engineers (ASME) so that users can apply both standards "in concert." We agree with the ANS External Events Working Group that, to achieve this consistency, the two standards must use identical definitions of terms.

The proposed ANS Standard avoids some of the weaknesses that we identified in our letters dated March 25, 1999 and July 20, 2000, concerning the ASME Standard. The ANS Standard provides an approach similar to that of Category II of the ASME Standard. It also provides guidance for seismic margin analyses that corresponds roughly to Category I of the ASME Standard. The ANS Standard, however, provides a good discussion of the limitations of these bounding analyses.

During the meeting, we offered a number of detailed comments on the Standard that the ANS representatives agreed to consider. We look forward to reviewing the proposed final ANS Standard following the reconciliation of public comments.

We commend the ANS External Events Working Group for the quality of this initial effort.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated January 24, 2001, from Shawn M. Coyne-Nalbach, American Nuclear Society, to Michael T. Markley, Advisory Committee on Reactor Safeguards, transmitting Draft BSR/ANS-58.21, "External Events PRA Methodology Standard" (December 25, 2000).
2. Letter dated July 20, 2000, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Final ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.
3. Letter dated March 25, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Phase 1).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 13, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C., 20555 - 0001

Dear Dr. Travers:

SUBJECT: REVIEW OF THE SIEMENS POWER CORPORATION S-RELAP5 CODE TO APPENDIX K SMALL-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSES

During the 479th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2001, we met with representatives of the Siemens Power Corporation (SPC) and the NRC staff to discuss the approval of the SPC S-RELAP5 thermal-hydraulic code for small-break loss-of-coolant accident (SBLOCA) analyses pursuant to the requirements of Appendix K to 10 CFR Part 50. Our Subcommittee on Thermal-Hydraulic Phenomena discussed this matter during meetings held on March 15 and August 8-9, 2000, and on January 16-17, 2001. We also had the benefit of the documents referenced.

Conclusion and Recommendation

1. We agree with the staff's decision to approve the use of S-RELAP5 to satisfy Appendix K requirements for analyses of the SBLOCA.

2. Documentation must be improved if it is to be suitable to support a "realistic" (best-estimate) version of the code. This applies to all features of the code, particularly the basic formulations, the solution procedure, assessment versus data, and the evaluation of uncertainties.

Discussion

The S-RELAP5 code has evolved from the earlier RELAP5/MOD2 and /MOD3 code versions, which have been used extensively and have been approved for SBLOCA analyses. SPC has made several significant improvements to the code, including better correlations, a more robust solution procedure, and a two-dimensional modeling option.

We agree with the staff's judgment that the code assessment performed by SPC is adequate for Appendix K SBLOCA analyses. In its original submission, SPC made comparisons with five tests. Two of these, the system tests Semiscale S-UT-8 and BETHSY 9.1b, posed substantial challenges that the code met successfully. In response to requests for additional information, SPC submitted further code calculations and data comparisons that increased confidence in the suitability of the code for the uses described in the Safety Evaluation Report (SER). These are the major reasons why we agree with the staff's decision to approve the code for Appendix K SBLOCA analyses.

Observations

Although we find that the staff has sufficient grounds to approve the present application of the code to Appendix K SBLOCA analyses, we have several observations that the staff should consider as it prepares to review the "realistic" (best-estimate) version of S-RELAP5.

1. SPC submitted documentation that had shortcomings of the type described in our report dated January 11, 2001. Although our Subcommittee had identified these shortcomings, there is little discussion of them in the staff's SER. During the August 8-9, 2000 Thermal-Hydraulic Phenomena Subcommittee meeting, SPC presented a different formulation of the mathematical models than that included in the written material submitted before the meeting. It is unclear to us what the final documentation will contain.

   Good documentation is sound quality control practice. It provides insurance against costly delays, uncertainties, confusion, and mistakes. It simplifies and enhances staff and ACRS reviews and aids users. It also builds confidence in the soundness of regulatory judgments in the broader technical community.

   In the future, the staff should insist on complete documentation before issuing a final SER.

2. The Appendix K provisions for code assessment for SBLOCA as listed in NUREG-0737 specifically identify only two comparisons with system tests: Semiscale Test S-07-10B and LOFT test L3-1. The SBLOCA covers a range of break sizes, leading to different rates of flow and a variety of conditions throughout the system. A more complete set of comparisons would increase confidence that the code has not been tuned to a single sequence of events and that some conditions where it might not perform appropriately remain undiscovered.

   The staff needs to consider how broad-based the assessment of realistic codes should be, not only to ensure adequacy but also to measure uncertainty. Clear criteria are also needed on what constitutes an adequate database for assessing this uncertainty and on how this should be done quantitatively.

3. In describing the Upper Plenum Test Facility (UPTF) test of loop seal performance, SPC stated [Page 5-46 of SPC Topical Report EMF-2328(P)] that in one case "the level predicted by S-RELAP5 was about 3.5 times greater than the measured level," while in another case "the predicted pressure drop across the loop seal was 10.9 kPa versus about 2.9 kPa from the data." SPC argued that these rather large deviations are conservative for the purpose of evaluating the success criterion of peak clad temperature during a SBLOCA. However, there are other applications of thermal-hydraulic codes for which deviations of this magnitude might prove to be unacceptable. For example, the performance of passive plant designs is likely to be more sensitive to the balance between hydrostatic driving forces and pressure drops throughout the system and may require greater accuracy in code predictions of these phenomena.

4. SPC provided the staff with a working version of their code and input decks to enable test conditions to be simulated. However, the staff informed us that it had not run the code as an independent check, nor used this capability to investigate some key features. We understand that the staff's rationale in this particular case is that it is familiar with previous relevant applications of RELAP5. The use of the codes by the staff should be an important part of its review process. We look forward to staff reports on its independent evaluation of code runs when S-RELAP5 is submitted as a realistic code.

5. Because we cannot check many features of a complex code, some of our assessment must be based on establishing confidence in the applicant's technical judgment. In the present case, we have been helped in our evaluation by the cooperation of SPC in responding to our technical questions and supplying additional information. Another important factor in establishing confidence is the provision of accurate, complete, and unequivocal documentation. We look forward to reviewing the revised documentation supporting the realistic version of the code.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Siemens Power Corporation Report, EMF-2100(P), Revision 2, "S-RELAP5 Models and Correlations Code Manual," January 2000 (Proprietary).
2. Siemens Power Corporation Report, EMF-2328(P), Revision 0, "PWR Small Break LOCA Evaluation Model, S-REALP5 Based," January 2000 (Proprietary).
3. Siemens Power Corporation Report, EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," November 1999 (Proprietary).

4. Siemens Power Corporation Report, EMF-2101(P), Revision 1, "S-RELAP5 Programmers Guide," December 1999 (Proprietary).
5. Siemens Power Corporation Report, EMF-CC-097(P), Revision 4, "S-RELAP5 Input Data Requirements," December 1999 (Proprietary).
6. U. S. Nuclear Regulatory Commission, Draft Safety Evaluation Report by the Office of Nuclear Reactor Regulation for EMF-2328(P), "PWR Small-Break LOCA Evaluation Model, S-RELAP5 Based," undated (Predecisional).
7. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "S-RELAP5 Request for Additional Information (RAI)," undated (Proprietary).
8. Report dated January 11, 2001, from D. A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Issues Associated with Industry-Developed Thermal-Hydraulic Codes.
9. U. S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 7, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.XXX, "FIRE PROTECTION FOR OPERATING NUCLEAR POWER PLANTS" (FORMERLY DG-1097)

During the 480th meeting of the Advisory Committee on Reactor Safeguards, March 1-3, 2001, the Committee considered the proposed final regulatory guide and decided not to review it. The Committee has no objection to issuing the final regulatory guide for industry use.

Reference:
Note dated February 14, 2001, from Eric Weiss, Office of Nuclear Reactor Regulation, to James E. Lyons, Advisory Committee on Reactor Safeguards, Subject: Request for Concurrence on Comprehensive Fire Protection Regulatory Guide

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
S. Collins, NRR
G. Holahan, NRR
E. Weiss, NRR
E. Connell, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 8, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT REPORT, "REGULATORY EFFECTIVENESS OF THE ANTICIPATED TRANSIENT WITHOUT SCRAM RULE"

During the 479th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2001, we met with representatives of the NRC staff to discuss the staff's draft report on the regulatory effectiveness of the anticipated transient without scram (ATWS) rule. We also discussed this matter during the 480th meeting, March 1-3, 2001. We had the benefit of the documents referenced.

We agree with the general conclusions of the draft report. As noted in our report on the Regulatory Effectiveness of the Station Blackout Rule, dated June 22, 2000, we believe the assessment of whether or not the regulatory analysis and the subsequent rulemaking are achieving the desired objectives to be an important part of the regulatory process.

The report notes the strong influence of the moderator temperature coefficient (MTC) on PWR ATWS performance and of operator actions on BWR ATWS performance. Industry initiatives to achieve longer cycles could result in insufficiently negative MTCs at full power for a larger fraction of the cycle length. Similarly, industry initiatives to uprate the power of existing BWRs could accelerate the progress of accident sequences, thus increasing the challenge to timely and appropriate operator action. Such consequences could erode the safety benefits derived from the implementation of the ATWS rule. We recommend that the approval of such fuel cycle changes and power uprates be contingent on maintaining the ATWS risk at acceptable levels.

Although this particular study answered a number of appropriate questions, we suggest that future studies of regulatory effectiveness explicitly address the following questions:

1. What contribution to risk was associated with the pertinent sequences before the rule was promulgated?

2. What level of uncertainty was attributed to the determination of the risk contribution?

3. In view of Items 1 and 2, why were these levels of risk and associated uncertainty considered unacceptable?

4. What were the target levels of risk and associated uncertainty?

5. Why were these target levels considered acceptable?

6. What plant changes were implemented as a result of the rule?

7. What reductions of risk and associated uncertainty were actually achieved by implementation of the rule?

8. What was the original regulatory analysis estimate of the cost of implementing the rule?

9. What was the actual cost associated with implementation of the rule?

We believe that answering these questions will lead to a more systematic and risk-informed assessment of regulatory effectiveness.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated October 18, 2000, from Farouk Eltawila, Office of Nuclear Regulatory Research, NRC, to David Modeen, Nuclear Energy Institute, transmitting Draft Report, "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," by Regulatory Effectiveness Assessment and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, NRC.
2. Memorandum dated July 19, 1983, from William J. Dircks, Executive Director for Operations, NRC, for The Commissioners, NRC, Subject: SECY-83-293, Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events.
3. Letter dated June 22, 2000 from Dana A. Powers, ACRS Chairman, to William D. Travers, Executive Director for Operations, NRC, Subject: Draft Report, "Regulatory Effectiveness of the Station Blackout Rule."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 15, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C., 20555-0001

Dear Dr. Travers:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE RETRAN-3D THERMAL-HYDRAULIC TRANSIENT ANALYSIS CODE

During the 480th meeting of the Advisory Committee on Reactor Safeguards, March 1-3, 2001, we discussed the status of the Committee's review of the Electric Power Research Institute (EPRI) RETRAN-3D thermal-hydraulic transient analysis code. Our Subcommittee on Thermal-Hydraulic Phenomena most recently discussed this matter with representatives of the NRC staff, EPRI, and its contractors during a meeting on February 20, 2001. We also had the benefit of the documents referenced.

In early 1999, we reviewed the RETRAN code documentation. On July 14,1999, ACRS Member Dr. Graham Wallis presented a critique of the momentum equations in RETRAN to the ACRS. During 1999 and 2000, the staff raised several questions concerning the momentum equations, both informally and formally, through requests for additional information (RAIs). EPRI responded to these RAIs on April 27, 1999, October 22, 1999, and March 6, 2000. Additional written material was submitted by EPRI on February 15, 2001. During the February 20, 2001, Subcommittee meeting, EPRI representatives agreed to reconsider the justifications of the momentum equations in RETRAN and the example problems illustrating their use for modeling specific components.

The major concerns identified by the Thermal-Hydraulic Phenomena Subcommittee regarding the momentum equations are summarized by ACRS Member Dr. Graham Wallis in the attached documents.

Sincerely,

George E. Apostolakis
Chairman

Attachments:
1. "Discussion on Momentum Equations," by ACRS Member Graham Wallis, dated February 25, 2001.
2. "Comments on EPRI Response to RAIs and Other Recent Submittals Concerning the RETRAN Code," by ACRS Member Graham Wallis, dated February 25, 2001.

References:

1. Safety Evaluation Report by the Office of Nuclear Reactor Regulation for EPRI NP-7450 "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," undated.
2. Letter dated February 15, 2001, from L. Agee, Electric Power Research Institute, to Graham Wallis, ACRS, Subject: Closure of the NRC RETRAN-3D Review.
3. Letter dated March 6, 2000, from G. Swindlehurst, Duke Power, to NRC Document Control Desk, Subject: Project No. 669 - Review of RETRAN-3D, Submittal of Additional Information.
4. Letter dated October 22, 1999, from G. Swindlehurst, Duke Power, to NRC Document Control Desk, Subject: Project No. 669 - Review of RETRAN-3D, Response to RAI Letter dated August 25, 1999.
5. Response to Office of Nuclear Reactor Regulation Request for Additional Information, EPRI Topical Report NP-7450, RETRAN-3D Project No. 669, dated April 27, 1999.
6. Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, "Staff Conference Call Follow Up," documenting results of NRR/EPRI conference call of August 29, 2000, pertaining to NRR review of RETRAN-3D.
7. Letter dated November 29, 2000, from Theodore Marston, EPRI, regarding ACRS Member Graham Wallis presentation to Commission regarding need for more realistic (best-estimate) thermal-hydraulic computer codes.
8. Comments by Graham Wallis on his review of RETRAN-3D, "Continuation of Review of RETRAN-3D," dated May 23, 1999.
9. Memorandum dated September 2, 1999, from P. Boehnert, ACRS, to ACRS Members, Subject: Supporting Documents - G. Wallis's Discussion on Status of EPRI RETRAN-3D Code Review - 465th ACRS Meeting, September 1, 1999.

DISCUSSION ON MOMENTUM EQUATIONS

By ACRS Member Graham Wallis, February 25, 2001

The momentum balance equation for a stationary control volume is (see any textbook)

$$d/dt \int (\rho\ v)\ dv = -\int p\ \mathbf{dA} + \int \boldsymbol{\tau} . \mathbf{dA} - \int \mathbf{v} (\rho\ \mathbf{v} . \mathbf{dA}) \quad (1)$$

For engineering purposes, this is usually reduced to the form

$$\mathbf{I}\ dW/dt = -\ \Sigma\ p_i\ \mathbf{A}_i + \mathbf{F}_w - \Sigma (\rho\ \mathbf{v}_i . \mathbf{A}_i)\ \mathbf{v}_i \quad (2)$$

Equation (2) is a node/port description where the velocities, v, at each port, i, are assumed to be uniform. The usual idea is to compute the rate of change in flow rate, dW/dt, across some internal surface in the node and step forward in time. The flow rates, W, throughout the system modeled by a set of such nodes will be solution variables that are updated as the numerical transient proceeds. The coefficient, **I**, is the effective vector inertia of the fluid in the node, with units of length. It represents an approximation, particularly if the flow is not uniform. It is also a significant assumption that the momentum in the node is proportional to the flow rate, W, (which is a scalar quantity) across some defined surface in the node. This is not so bad for single phase incompressible flow with ports at the end of the nodal volume, because the flow is the same across any surface in the node that does not intersect the ports. For more general compressible or multiphase flows with many ports, the momentum in a nodal volume is not so easy to figure out. $\mathbf{F}_w$ is the force from the walls. The shear stress contribution to the forces at the ports is usually neglected.

To illustrate the importance of the wall force, consider a couple of parallel similar pipes in the x-direction joined by a 180-degree bend in the horizontal x-y plane and filled with an incompressible inviscid fluid. The momentum in the two pipes cancels and the total momentum in the system is all in the y-direction. The pressure and momentum flux terms on the right hand side of Equation (2) all act in the x-direction, so it is only the net wall force acting in the y-direction that is available to change the net fluid momentum in the system. This force may actually be computed by first

ATTACHMENT 1

using mechanical energy conservation to get the acceleration and then using the y-component of the momentum balance to deduce the wall force.

There are several important features of Equation (2) that present difficulties to the code developer:

1. It is a vector equation. It can only be reduced to one-dimensional form if the flows and forces all act in a single direction, which is not the case for flow around a bend, for instance. If it is resolved in some direction to obtain a scalar component, then all terms must be resolved in a consistent way.

2. The force from the walls is unknown and cannot be determined from known quantities without invoking some new information, except in trivial cases which are probably limited to a straight pipe. This force is made up of resultants from both normal (pressure) and tangential (shear) components.

3. The pressures at the ports or junctions (node boundaries) act on areas. These areas cannot be made to disappear except when the flow is in a straight pipe and the equation can be divided through by the area. No amount of algebra can make the areas disappear in the general case, though the "momentum" equations in some codes are written without areas multiplying pressures at the junctions. To get an equation like Bernoulli's in which the pressures do not multiply areas and the formulation is one-dimensional, you have to integrate a differential form of the momentum balance along a streamline. This is strictly invalid when streamlines get mixed up in nodes, through turbulence or flow separation, but such an approach has also been tried as an alternative way to get usable equations for codes.

The biggest problem is Item 2. It is basically insurmountable in any general way. Attempts have been made to derive the force from the walls from another principle, such as conservation of mechanical energy. However, the forces from walls are usually imposed by stationary surfaces. They, therefore, do not work and do not contribute to the energy balance. Therefore, there is no way that the energy balance can be manipulated to solve for the wall force.

Conservation of mechanical energy is sometimes used in place of the momentum balance to provide an expression for dW/dt. Bird, Stewart, and Lightfoot [Reference 1] discuss the conditions for validity of such a balance (e.g., constant density or constant temperature). They solve the example of oscillations of a manometer this way. This method has not been developed as a general derivation that might apply to two-phase flows of the type that occur in reactor systems.

The approach taken in all codes is to derive a momentum balance for an extremely simple geometry, such as a long straight pipe. The result is then usually applied with little or no explanation or justification to other shapes and situations. I think it is true that the (long) straight pipe is the only case in which it is possible to overcome the three difficulties listed previously. With some allowances for "averaging," Equation (2) can then be expressed as

$$L\, dW/dt = p_1 A - p_2 A - \tau_w \pi D L + \rho_1 v_1^2 A - \rho_2 v_2^2 A \qquad (3)$$

Where L is the length of the pipe, subscripts denote the inlet and exit; A is the cross-sectional area, D the diameter (or effective diameter), and the velocities are all in the direction of the pipe axis. The wall shear stress is computed from the steady-flow friction factor, though friction is strictly not the same in unsteady flow. If Equation (3) is divided by A and the pipe is assumed to be circular, we get

$$L/A\ \ dW/dt = p_1 - p_2 - \tau_w 4L/D + \rho_1 v_1^2 - \rho_2 v_2^2 \qquad (4)$$

If the fluid is incompressible or suffers no change of density, the last two terms cancel each other and disappear. Similar equations can be deduced for each phase in the two-fluid model.

Even when applying these methods to straight pipes, care may need to be taken near ends or junctions where flow is not one-dimensional. The lengths, L, of nodes must be chosen to correspond to regions where the properties do not change too rapidly.

It is not directly evident from the documentation, but presentations from proponents of the RELAP and TRAC codes lead me to conclude that most of the reactor system is modeled as a series of straight pipes connected by nodes of zero length that contribute frictional losses but no inertia. Bends, for example, are modeled as a series of these straight pipe segments and the additional losses contributed by the non-straight shape are added in between these segments. More complex nodes are modeled in an *ad hoc* manner that has evolved with time and experience.

RELAP and RETRAN also make use of a derivation for two coaxial straight pipes connected by a sudden change of area. The pressure difference across the junction is taken as given by the steady flow loss coefficient and it is assumed that this all occurs in zero length. This is no different from the idea of joining two straight pipes with a valve or other "resistance" and there is no need for the pipes to be oriented in the same direction.

Denoting one pipe by the subscript "a" and the other by "b" we have two equations like Equation (4) as follows:

$$L_a/A_a \; dW_a/dt = p_1 - p_2 - \tau_w 4L_a/D_a + \rho_1 v_1^2 - \rho_2 v_2^2 \qquad (5)$$

$$L_b/A_b \; dW_b/dt = p_3 - p_4 - \tau_w 4L_b/D_b + \rho_3 v_3^2 - \rho_4 v_4^2 \qquad (6)$$

The pressure change across the junction is assumed to be given by the steady-state correlation, which could take the form,

$$p_2 - p_3 = k \; ½ \; \rho_2 v_2^2 \qquad (7)$$

with "k" being a loss coefficient for the junction.

There is nothing special going on here, just building up a composite piece of a circuit by combining two straight pipes and a junction.

Having read Bird, Stewart, and Lightfoot, the RELAP developers decided to express the empirical losses across the junction another way. The pressure change is expressed in terms of mechanical energy losses, or as a loss in Bernoulli head. This is strictly only valid for an incompressible fluid, though some workable derivations might be possible for other conditions, such as isothermal flow, if done carefully. Then Equation (7) is expressed as

$$p_2 - p_3 = -½ \; \rho_2 v_2^2 + ½ \; \rho_3 v_3^2 - k_e \; ½ \; \rho v^2 \qquad (8)$$

where k_e is a coefficient of mechanical energy loss. I have left the velocity and density in the last term without subscripts as the appropriate conditions have to be defined. This, of course, is part of the definition of the empirical loss coefficient k_e. I have also used subscripts on the "kinetic energy" terms, adding to the definition of the loss coefficient. I believe this loss coefficient is simply taken from single-phase flow tests, so it is something of a reach to apply it to an unsteady two-phase flow with density change.

If we use Equation (8) to eliminate the intermediate pressures, p_2 and p_3, from Equation (6) plus Equation (7) the result is

$$L_a/A_a \; dW_a/dt + L_b/A_b \; dW_b/dt = p_1 - p_4 + (- \tau_w 4L_a/D_a - \tau_w 4L_b/D_b - \tfrac{1}{2} \rho_2 v_2^2 + \tfrac{1}{2} \rho_3 v_3^2 - k_e \tfrac{1}{2} \rho v^2) + \rho_1 v_1^2 - \rho_4 v_4^2 \quad (9)$$

In RETRAN, it is asserted that the two terms on the left hand side can be combined by assuming that both of the W's are the same as some "W" for the "junction". The term in parentheses is interpreted as some sort of total loss for the system, and the two last terms are interpreted as momentum fluxes in and out of the combined system. This is how the A's are made to disappear from what would be an equation resembling Equation (2) if written as the momentum equation for the whole works of two pipes plus junction. It then seems to be assumed, without argument, that a similar equation applies to any shape or component in the system, except when a special model is derived, as for a pump. RETRAN has sketches of more general shapes, but there is no proper derivation of a momentum balance for them, just an equation written down to look like the "two-pipe-plus-junction" (TP+J) case.

Note that Equation (9) is a scalar equation, unlike Equation (2). It does not represent a "momentum balance" for a control volume and it cannot be "resolved" in some direction. However, in RETRAN a modification is made to change the two final terms in Equation (9) to $\rho_1 v_1 v_{1,\psi} - \rho_4 v_4 v_{4,\psi}$ where the subscript is supposed to denote the "component that lies in the direction of the junction". I have yet to see a convincing derivation of this result. It seems to be a sort of hybrid between Equation (2) and Equation (9) in which the momentum flux terms are resolved in some chosen direction, as the ones in Equation (2) would have to be to obtain a scalar result. Since the direction is arbitrary, different results can be achieved, over a limited range, at the will of the user.

If the fluid is incompressible, or of constant density, then $v_1=v_2$ and $v_3=v_4$ so that Equation (9) reduces to a form of Bernoulli equation with losses (which the RETRAN version with "resolved" momentum fluxes does not, an indication that something is almost certainly wrong). This particular result can be deduced from the principle of mechanical energy conservation, as long as the density is constant, which is not the case in a general two-phase flow.

The TP+J model can also handle some aspects of momentum addition from side branches, as in ECC injection into a cold leg. If a flow W_{sa} is injected from a connection to the side of pipe "a" with velocity component v_{sa} in the direction of the pipe axis, then an additional source of momentum equal to $W_{sa}v_{sa}$ appears on the right hand sides of Equations (5) and (9). RETRAN also has such a term, but the definition of the velocity component is ambivalent. The example of the wye-junction in the RETRAN text (EPRI NP-1415 [Reference 2]) seems to indicate that this term was improperly evaluated in that case.

The RETRAN documentation at least acknowledges that there is a need to develop an equation describing a general shape with several connections to ports or junctions. There is just no good rationale for the result and no examples showing how to use the method for the sorts of nodes, other than straight pipes, that occur in models of nuclear systems. There are some other concerns with the documentation, including:

1. Derivations of momentum equations in various forms that appear questionable.

2. Examples of applications to bends, tee-junctions, wye-junctions that appear wrong at an elementary level, even if one accepts the basic equation used.

3. Strange features, such as resolving the scalar flow rate in each coordinate direction as if it were a vector and interpolating these components in ways that seem to defy physical reality. This shows up also in the worked examples, where some odd terms are derived.

4. Misplaced appearance of rigor, when it would be better to explain and justify assumptions.

5. A method of "resolving" the momentum flux terms that seems to be arbitrary and makes it possible to achieve a range of different results, depending on the user's choice of the angle ψ.

These points are examined in more detail in the accompanying document "Comments on EPRI Response to RAIs and other Recent Submittals concerning the RETRAN code."

Do these inadequacies or limitations or "assumptions" matter for the purposes of nuclear safety calculations? Perhaps. In some cases, the transients are so slow that the momentum balance collapses to the steady flow result and correlations for "pressure drop" suffice. Some transients appear to be dominated by the mass and energy balances, which are much easier to compute, as they deal with scalar quantities and the transfer from walls can be evaluated. In other cases, things may not be so simple. Because all the treatments of momentum balances are very rough approximations, it would seem a good idea to run sensitivity tests on all the coefficients, and perhaps on the structure itself, in these equations to explore if and when this makes any significant difference to safety conclusions and to provide explicit guidance for a user about possible problems or limitations.

In any case, it is not good for public confidence to have documentation that appears of doubtful validity to an informed observer.

Nomenclature:

A area
D diameter
F force
k loss coefficient
L length
p pressure
t time
v velocity
W mass flow rate

ρ density
τ shear stress
ψ angle defined in RETRAN

Bold symbols denote vectors or tensors

Subscripts:

a, b Two pipes
e energy
i a general port or junction
s from a side junction
w at the wall
1,2 ends of the first pipe
2,3 before and after the junction
3,4 ends of the second pipe

References:

1. R. B. Bird, W. E. Stewart, and E. N. Lightfoot, "Transport Phenomena, John Wiley & Sons, New York, NY, 1976.
2. G. F. Niederauer, C. E. Peterson, E. D. Hughes, and W. G. Choe, "Application of RETRAN to Complex Geometries: Two-Dimensional Hydraulic Calculations," EPRI NP-1415, 1980.

COMMENTS ON EPRI RESPONSE TO RAIs AND OTHER RECENT SUBMITTALS CONCERNING THE RETRAN CODE

By ACRS Member Graham Wallis, February 25, 2001

ACRS reviewed the documentation of the RETRAN code in early 1999. On July 14, 1999, Dr. Wallis presented a critique of the momentum equations in RETRAN to the ACRS. During 1999 and 2000 the staff raised several questions concerning the momentum equations, both informally and as formal requests for additional information (RAIs). EPRI submitted responses to these RAIs on April 27 and October 22, 1999 and March 6, 2000. Additional written material was submitted by EPRI on February 15, 2001. On February 20, 2001 representatives of EPRI and their contractors met with the ACRS Subcommittee on Thermal-Hydraulic Phenomena at NRC headquarters in White Flint. At this meeting, EPRI agreed to reconsider the justifications of the momentum equations in RETRAN as well as the example problems illustrating their use for modeling specific components.

This document has been prepared to assist EPRI in identifying the major concerns of the ACRS and to facilitate their response. Since the uses of the momentum equations are pervasive in RETRAN, it is likely that some illustrations and derivations, resembling those cited in this report, have not been specifically identified. EPRI should therefore ensure that any proposed modifications or corrections to the RETRAN documentation and/or code content are comprehensively and consistently applied in any new versions.

Reference is made to the accompanying "Discussion on the Momentum Equations" prepared by Dr. Wallis.

REVISED DOCUMENTATION SUBMITTED WITH RAI RESPONSES

EPRI enclosed "Revision 5" [Reference 1] of their RETRAN documentation. The momentum equations are described in Section 3.

Figure II.3-1 shows a straight pipe, about which there is little disagreement.

Figure II.3-2 shows a bend. It is described as "a slight generalization." The bend looks rather gentle, but there is nothing in the text that says that the angle through which the flow is turned is small. No approximations seem to be made assuming that the angle is small, so it appears that the method should apply to any bend, including a 180 degree one, for example. Section 3.1.2.1 is entitled "Constant Area Channels," yet the equations retain different areas A_k and A_{k+1} which appear later in the supposedly more general form Equation (II.3-27) which is written down with no additional explanation.

Equation (II.3-4) is the vector momentum balance. It should contain the resultant forces from normal and tangential stresses at the wall. Reference is made to Equation (II.2-34) to explain how the wall forces are divided up, but this equation (in Revision 1[Reference 2], which is what we have as the original basic document) only gives a very general form and does not explain the three terms appearing in Equation (II.3-4). F_{loc} later gets called the "form losses". It is presumably the resultant of normal stress components, because it gets combined with the surface pressures on the fluid surfaces later down the page. This combination does not help, as the components are later separated again.

"Assuming a uniform pressure along the surface within each region" to get Equation (II.3-7) is not useful because it throws out the important physics. If the fluid were subjected to uniform pressure, there would be no resultant force from that source. Even if true, it would not lead to the disappearance of the wall force due to normal stresses. In steady flow around a bend, the wall reaction is the force that turns the flow and enables the exit momentum to be in a different direction from the inlet momentum. This is especially evident for a 90 degree or 180 degree bend. When the flow accelerates, as in a transient, the wall force must also be considered. It is the only force providing the y-momentum change for a horizontal 180 degree bend with end faces in the x-direction, for example.

Equation (II.3-7) appears to be the component of a momentum conservation equation in the direction "i" and ψ is the angle between the directions k and i. The momentum fluxes are resolved in this direction. None of the friction forces, the gravitational forces or the pressure forces are resolved in this direction, therefore this cannot be the scalar component of a vector equation. Also, if this were based on a vector equation, the inertia terms on the left-hand side would have to be resolved in the chosen direction, so that the L's appearing in Equation (II.3-9) would have to be projected in that direction or redefined somehow.

The momentum flux terms contain different areas with subscripts k and k+1. The pressure terms do not. This is either an inconsistency or a sign of conceptual confusion.

The resultant of normal forces from the walls is omitted, though playing a key role in all bends that turn a flow through a significant angle.

The equation at the bottom of the page defines "a component of the volume centered flow." Now, W is a scalar and does not have components. It is possible to define a variable by using the form at the bottom of the page, but it has to be used very carefully, as it has no direct physical interpretation and may well mislead (or itself be a symptom of misunderstanding).

(Many of these points were brought up in previous ACRS critiques of this work.)

Section 3.1.2.2 is entitled "Variable Area Channels." Figure II.3-3 actually shows a very specific shape. It is analyzed in its one-dimensional form rather like the TP+J model discussed in the "Momentum Discussion," though the figure should show two long pipes for this to be at all a good approximation. Equation (II.3-12) differs from the TP+J model in that the exiting momentum is resolved in the (mysterious) direction ψ which does not appear in the figure and should not be there if this is really a TP+J model. If this is supposed to be a momentum balance then all other terms, such as the pressure forces on the ends, must be resolved in this direction too. The gravitational terms should be resolved in appropriate directions along the pipe axes, and they are not, even if this is to be a TP+J model. This is another inconsistency. The equation is neither a true momentum balance nor representative of a true TP+J model but some sort of unjustified hybrid. The same is true of Equation (II.3-20), which is the more usual form of the RETRAN equation, containing those unusual "resolved" flow rates.

The idealization shown in Figure II.3-5 to represent "any junction" is so abstract and unexplained that it is hard to tell why it should be useful or how to use it without reference to worked examples. It seems unlikely that all configurations of interest can be forced into such a framework. There seems to be a leap of faith required to use Equation (II.3-27), which is merely a repetition of Equation (II.3-26).

It is stated that flow velocities are not necessarily normal to junctions, but have angles ϕ to them. This leads to discussions on Pages II-84 and II-85 of "Revision 5" in which the flow rates seem to be treated as vectors, which is unphysical. Figure II.3-5 is drawn with the end faces parallel to each other and normal to the direction "i" which seems to be defined by the junction around the middle of the picture. Are these features requirements of the model? What happens with less one-dimensional shapes? This figure is vague, and there is no derivation of the momentum equation for it, so there is really no way to check the validity of the result without looking at specific examples. However, it is probable that the momentum balance for a general control volume cannot always be idealized realistically in some arbitrary way like this.

Tee Example

The noding in Revision 5 is quite different from that in Revision 1. Does this mean that the "rules" for noding have changed in the code? How sensitive are the answers to the actual noding employed?

Equation (II.3-35a) is the x-direction momentum balance for the shaded volume in Figure II.3-7a. The contribution of W_4 in taking x-momentum out of the volume is ignored, though significant in reality, presumably because this flow is assumed to be all in the y-direction.

It seems to be being assumed that the zetas in Equation (II.3-28) are each 1/2. $W_{1,x}$(bar) (I can't figure out how to put a bar on the variable using this computer program, so I'll have to write them in) is set equal to $(W_1 + W_2)/2$. Because some flow is diverted to the side branch, it seems better to use $(W_1 + W_2 + W_4)/2$.

The use of $W_{1,y}$(bar) requires explanation as the flow appears to be perpendicular to the left-hand boundary of the control volume and not to have a y-component. Making it equal to $W_4/2$ is arbitrary and appears dubious. If one is going to reason this way, it should be considered that if only one half of W_4 comes in through the surface 1 (circled) then the other half must come in through the surface labeled 2 (circled) which is unlikely as flow is going out that way.

The arbitrary appeal to "applying the assumptions of steady-state conditions" is odd since the whole point is to develop methods for transients. Even more confusing is the expression for "volume centered flow" at the bottom of the page. It doesn't appear later, but would it somehow be used in the transient term in the momentum balance if this were to be shown in Equation (II.3-35b)?

Since $A_1=A_2$, there is no need for two areas in Equation (II.3-35b). The loss term is presumably quite small, if evaluated for the steady flow going straight through from 1 to 2. If some flow goes out the side branch, then it will influence the losses. Then, e_2* must depend on W_4.

The momentum flux term for area 1 is incorrect in Equation (II.3-35b). If $W_2=0$ and flow is steady, then $W_1 = W_4$. The flow coming into the control volume is W_1; therefore, the first momentum flux term should not have the 4 in the denominator. This correction would make $p_2 = p_1 + W_1^2/\rho_1 A_1^2$. But this answer defies Bernoulli's equation, if the fluid is inviscid and incompressible, which states that the maximum pressure rise is one half of this at the stagnation point somewhere on surface 2. The average pressure at 2 must be less than this maximum pressure. In reality a significant x-direction momentum is carried out of the cell by the flow W_4, reducing the predicted pressure rise at 2 to reasonable values. This important physical mechanism is ignored in Equation (II.3-35b)

The sign of the term in square brackets in Equation (II.3-35a) and Equation (II.3-35b) is the opposite of what it is in the original general Equation, Equation (II.3-26).

Equation (II.3-36a) is odd. It cannot be the y-component of a momentum balance because the pressure acting on surface 1 is in the x-direction while that on surface 4 acts in the y-direction. The subscript ψ is supposed to signify the component in some specified direction (here unspecified). If ψ is y, as implied, then we should be multiplying W_1 by $W_{1,y}$ in the first momentum flux term and not getting a factor of 4 in the denominator in Equation (II.3-36b) but a factor of 2. The second momentum flux term does seem to correspond to a y-direction flux, but it is unclear why the "assumption of steady-state conditions" can be used in a transient.

The sign of the term in square brackets in Equation (II.3-36a) and in Equation (II.3-36b) is wrong. The area A_2 in the square brackets in Equation (II.3-36b) should be A_1.

If this were a real momentum equation in the y-direction, p_1 would not appear, but the forces on the bottom and top walls in the y-direction would have to be evaluated. There is also flow out of the 2 face; presumably it is assumed to carry no y-momentum, though the flow across the 1 face was assumed to have this capability.

This Equation cannot be an example of the TP+J approach because the control volume has three connections to the outside world and cannot be modeled by two pipes. In any case, the pipes are not "long" by any means, and that is the condition needed for this approximation to be good.

It is actually not easy to derive a valid transient motion equation for this control volume. It cannot be analyzed using the overall "momentum equation" because of wall forces, and it does not conform to a simplified model, such as the TP+J case. It really needs to be modeled by some special method, such as running a CFD code and/or conducting experiments and fitting the results for a range of flow splits (main branch versus tee-branch) with an empirical "three-port" model. However, this does not excuse what appear to be conceptual errors in the RETRAN documentation.

Elbow Example

At the bottom of Page II-91, the "steady-state assumption" appears to be being used. This obscures the understanding of how the method is to be used to represent a transient. It would help if Equation (II.3-37b) included the transient term so that we could see how it is to be evaluated (e.g., what L's and W's are to be used). This is not clear from any description in the text.

This solution has changed from the previous version in Revision 1. In that case, the second momentum flux term was evaluated as the square of W_{2x} so that the factor in the denominator in Equation (II.3-37c) was 4 and not $2\sqrt{2}$. Neither version reflects the physics. If this is a TP+J model (how does that work for a bend?), then the factor should be 1. If it is a momentum balance in the x-direction, then the total flow, W_2, should be multiplied by the velocity component in the x-direction, giving a factor of $\sqrt{2}$ in the denominator. In this latter case, the pressure force on the surface 2 would have to be resolved in the x-direction and the reaction from the wall somehow determined and resolved in the x-direction too.

The "flow rates in the x- and y- directions" in the middle of Page II-93 appear contrary to any physical interpretation. If some sort of numerical interpolation is going on, it does not seem to correspond even to the simple situation in which the flow rate in the pipe is constant, as in steady flow. The "magnitude of the volume-averaged flow" likewise cannot be $1/\sqrt{2}$ times the steady-state flow and there is no reason to make this the case in unsteady flow either.

Why are A_1 and A_2 being retained when the pipe has a constant cross-section? If it does not, then the pressure forces need to be multiplied by different areas if a true momentum balance is being performed.

If Equation (II.3-37c) is evaluated for constant area and steady frictionless flow, it turns out that there is an artificial pressure recovery in the bend because the first term on the right-hand side is bigger than the second. One would expect the pressure to stay constant. During the February 20, 2001 meeting, EPRI claimed that this did not matter much as this pressure recovery was canceled out by the pressure loss in the second half of the bend. This is not necessarily so. If the angle ψ for the second part of the bend is chosen in the same way as for the first part of the bend, being in the direction of the inlet face, then the same artificial pressure recovery occurs. In a coil of several 360-degree bends, this pressure could be used to build up as much pressure as desired and create a "pump" with no energy input.

In the previous paragraph, it was shown that the answer depended on the choice of the arbitrary angle ψ. This appears to be a general fault with the "vector" RETRAN momentum equation. One can change the momentum flux terms, without changing anything else in the equation, just by changing ψ and resolving them in a chosen direction. For frictionless steady flow in a bend, for example, the pressure difference can be made to take any value between some positive and negative limits, depending on the user's choice of ψ. This is a very undesirable feature of what should be a deterministic method.

Wye-Junction Example

Dr. Wallis' presentation to the ACRS in 1999 also included similar critiques of the way in which the wye-junction was analyzed in EPRI NP-1415 [Reference 3], which is the twenty-year old report out of which the present RETRAN documentation evolved. The conceptual problems appear similar to those described above, though more extensive, partly because of the "cross-momentum" effects when flow crossing a surface introduces or removes momentum with a component in a direction parallel to the surface. If the documentation is to be modified to respond to the above points, then that example should probably also be corrected.

The Porsching Paper (The "old" one, dated October 15, 1999 [Reference 4], that came with the RAI responses)

This paper appears to be an attempt to justify the form of the RETRAN equation, such as Equation (II.3-26), apart from the "loss" terms.

Perhaps the first thing to note is that Porsching's Equation (10) is not compatible with Equation (II.3-26). Equation (10) is a momentum balance for the control volume, whereas the RETRAN equation is not. Dividing Equation (10) by A_o we find that the momentum flux terms have A_1A_0 and A_2A_0 in their denominators and not A_1^2 and A_2^2 as in Equation (II.3-26). The latter resembles the TP+J form, except for the (inappropriate) resolution of the momentum flux terms in the direction ψ. The RETRAN momentum flux terms are neither correct from the TP+J viewpoint nor from the "momentum balance resolved in a chosen direction" viewpoint. They are an invalid hybrid form.

The momentum flux terms in Equation (II.3-10) only have the same denominators because for this example all the areas are the same. The form in Equation (II.3-26) and Equation (II.3-27) has no physical basis, nor is one provided in the text.

Porsching's Equation (4) is acceptable if one is careful about the integration that enables the volume integral of momentum to be expressed in terms of an average flow rate across slices perpendicular to $\mathbf{n}_0$ throughout the volume. This is not spelled out in the paper. If the flow is incompressible or steady and the ends S_1 and S_2 are parallel to S_0, W_0 can be related to the flow rate across the particular surface S_0, but this is probably not possible in general. It is not correct that L_0 in Equation (4) is equal to V_Ω/A_0 for the incompressible or steady flow cases. It should be equal to the physical distance between S_1 and S_2 in the "0" direction, if the ends are perpendicular to this direction. Otherwise there are corrections for the pieces of volume that involve partial slices parallel to "0" that intersect the end faces. In a compressible or multiphase flow, it is quite possible for the flow rate across other surfaces in the volume to be unrelated to that across S_0 so that L_0 in Equation (4) becomes a variable that is dependent on all the details of the flow. In any case, something like Equation (4) may be acceptable as an engineering approximation if careful definitions and restrictions are specified.

Porsching's Equation (5) is also in the form of a common engineering approximation. The final step in that equation is not exact, any more than the square of an average value of something is equal to the average of the square of something. This is well known in fluid mechanics and is the basis of correction factors for the momentum flux in a pipe with a velocity profile, for example. However, Equation (5) and the resulting Equation (6) are usually acceptable as engineering approximations which might require reevaluation if the velocity profiles are far from uniform.

The major error, or at least misleading derivation, in the Porsching paper concerns the pressure term in Equation (10). The integrals in Equation (7) are over all the areas of surfaces to the left and right of S_0. They include the walls of the duct as well as the areas for flow, S_1 and S_2. It is usual to separate out the net pressure forces on the flow areas, i.e., the ports or junctions connecting to other volumes, and the net pressure force on the walls. Porsching's mathematics in Equation (7) defines p_1 as the average pressure on components of surface in the "0" direction over both the area S_1 and all the area of duct walls on the left hand side of S_0. Physically, this has the effect of combining the forces on the fluid area and on the wall area into one average pressure times a reference area A_0. The quantities p_1 and p_2 used in RETRAN are averages over the fluid areas alone and are quite different from Porsching's average pressures in his Equation (8). Similarly, the pressures used by Bird, Stewart, and Lightfoot [Reference 5] in Porsching's Equation (13) are averages over the fluid areas and are quite different from those in Equation (8).

<u>New Material Submitted by EPRI on 2/15/01</u>

This consists of a letter from Lance Agee, a "new" paper by Porsching [Reference 6] (dated April 18, 2000), and a further revision (5b) to part of the RETRAN documentation. The letter claims that the concerns were suitably addressed in the RAI responses and by the Porsching papers. As mentioned above, they do not remove ACRS concerns and rather serve to reinforce previous conclusions.

The new version of the documentation addresses the momentum equation for a bend, illustrated in Figure II.3-2. There is nothing about the bend being slight. Indeed the method is later applied to a 90-degree bend.

Here, for the first time, the authors consider the resultant of normal forces on the wall. [Actually also friction, if one wants to be exact. It is not true that the net friction and form forces are all taken care of by the steady-state pressure gradient, as claimed. A proper momentum balance for a general shape in steady flow will show that the net frictional force on the wall and the normal stresses associated with "form losses" do not just "balance the pressure difference" because the end forces have to be multiplied by the corresponding areas and resolved like vectors, while the pressure change does not. This is part of the continuing confusion in RETRAN between a true momentum balance and a "pressure difference" that crops up in a Bernoulli-like or "mechanical energy" or TP+J equation. To demonstrate this, consider a 180-degree bend of constant cross-section, with an incompressible fluid flowing through it in steady flow. The resultant of the wall shear stresses is in the diametral direction (0 degrees to180 degrees) while the pressure forces on the ends reinforce each other (rather than being in opposition) and act in the 90-degree direction, being balanced by the wall forces in that direction. The idea that friction forces and form losses balance pressure drop in the momentum equation is naïve and based on extrapolation of experience with a straight pipe].

There is an S_{tot} on the integral in Equation (II.3-5). Equation (II.3-5a) breaks this down into forces from the end faces and from the walls. Equation (II.3-6a) is similar to the derivation in the "old" Porsching paper. In this equation, the p_k and p_{k+1} are not average pressures over the junctions but are averages over the entire surface of the control volume including the walls. They are quite different from the average pressures over the ends. The math from Equation (II.3-6b) to Equation (II.3-6e) is essentially the same as was used by Porsching ("old" paper), except that in his more general case, the A in Equation (II.3-6e) would have the subscript 0. Equation (II.3-7) is essentially Porsching's Equation (10) with no allowance for the different subscripts on the areas, which confuses its later modification to a form in which the areas of the inlet and outlet and some characteristic area (A_0) of the volume are all different. (The earlier version of this derivation, Revision 1, contained an upstream area A_k and a downstream area A_{k+1}. These multiplied the corresponding pressures in the momentum balance, Equation (II.3-9) but were not resolved in the direction ψ. These area factors were made to disappear in Equation (II.3-10) of Revision 1, the "RETRAN equation," by making the areas equal and dividing the equation by the area. When the areas are unequal this cannot be done and the RETRAN equation does not result. It is even stated in Revision 1 that Equation "(II.3-10) is valid only for the case of flow in a channel of constant cross-sectional area."

At the presentation on February 20, an argument was advanced that the pressures could be assumed to be uniform in the two regions before and after the "junction." In this case, there is no need to perform the integrations between Equation (II.3-5a) and Equation (II.3-6e). ACRS consultants opined that such sweeping assumptions in effect throw out the major physical phenomena and should not be made. In a more mathematical sense, there is no direct relationship between the average pressure over a volume and the average pressure over the surface area surrounding that volume. As an example, the force from the walls that turns the flow in a bend reflects the difference in the pressure forces on the inner and outer sides of the bend. If one applies the volume-average pressure over the whole surface, there is no force to prevent the flow from continuing straight ahead.

In sum, the critique of Porsching's "old" paper outlined above appears to apply equally well to the newest attempt to justify the RETRAN equation, albeit in a simplified form. Average pressures of various sorts should not be mixed up. There is also a sleight of hand in deriving a result in which all areas are equal and later generalizing it to cases where they are not.

The new Porsching paper (April 18, 2000) appears to recognize two of the basic problems outlined in the "Discussion," but his resolution of them seems inconclusive, merely suggesting that some sort of engineering approximation might be found.

His Option 1 is the old story. Equation (19) is the former Equation (10) with all the previous faults. The pressures appearing there are averages over the entire surface and not just over the ports or ends.

Option 2 is a new variation that appears essentially the same, but seems to involve resolving the total areas on each side of A_0 into two arbitrary directions. It is not clear how this helps to get rid of the net force from the wall (it is physically real and cannot be excluded from a macroscopic momentum balance by mathematical juggling).

It is unclear if there is a problem with the orientation of surfaces, as discussed under "Remarks." The area A_0 is equal to the area of any closed surface built on it, to the right or left, as long as one keeps track of the vector nature of surface elements. These surfaces can have any number of folds and wrinkles. That is not the problem.

Equation (26) seems to face up to the real problem. The total pressure force on one side is made up of the contribution from the walls and that from the end. The average pressure on the end is defined in Equation (27) as p_1 with a bar on it, recognizing that it is distinct from the p_1 that appeared in Equation (19). The effort now becomes to make the wall force, the last term in Equation (28), go away somehow. This is acceptable for a straight pipe [Case (a)], and perhaps as an approximation for a pipe with a slight bend or wrinkle in it [Case (b)]. But there is no justification for neglecting the term in general and none seems to be offered.

Section 2 of the "Remarks" admits another fundamental problem, how to relate the various W's to each other. However, there appears to be nothing definite in this section that resolves the problem, just a discussion of how "averaging' might be the way to do it.

<u>RAI 1</u>

This refers to Attachment 2 and is concerned with explaining how the RETRAN momentum equation applies to nodes of more complex shapes.

Figure 1 shows a straight pipe and is useful for defining the staggered grid approach and nomenclature.

Equation (3a) is said to be the "one-dimensional mixture momentum equation." As it involves two different areas, it cannot be a momentum balance equation because the pressure terms in Equation (3a) do not multiply areas. It must apply to a different shape than in Figure 1, probably a tapered pipe or two pipes joined together. It resembles Equation (9) in the "Discussion," the "two-pipe-plus-junction" model (TP+J), yet does not contain the $1/2\ \rho\ v^2$ terms and does not reduce to Bernoulli's equation (as it must) when there is no friction. So, this seems to be an equation that does not conform to any known pattern.

On page 5 (about the middle of the page), there is mention of "the component of the volume average flow which lies in the direction of the momentum cell." Now, there is no component of a scalar quantity like W, so it is unclear what this means. It is also uncertain what the "direction of a momentum cell" is when it has multiple inlets and exits or a complex shape.

The shapes shown in Figure 2 should be very useful for checking what the RETRAN momentum approach actually implies. "Junction 2 Cold Leg to Downcomer" is a bend, a classical sticking point for use of momentum conservation. Equation (5) is to be applied. It more closely corresponds to the TP+J model mentioned in the "Discussion" but (only) the momentum flux terms are resolved in a chosen direction. $W_{k,\psi}$ is said to be the "component (of the flow) that lies in the direction of the junction." As W is a scalar, it is unclear what this means and one has to look at the examples to figure out how to interpret the concept.

Tables 1 and 2 are intended to explain things. From Equation (3) and Equation (5), it appears that the W's with bars over them describe the flows at the boundaries of the momentum cells and the W's without bars are the flow rates in the cells that are part of the inertia term on the left hand side of the "momentum equation." What is meant by a "junction" is less clear, since the momentum and mass cells have different (staggered) boundaries. It looks as if the idea is that the numbers without circles on them in Figure 2 label "junctions" while the circled numbers label "volumes," so these must be the mass and energy nodes that are being described. (It looks as if the 1 above the cold leg in the lower figure should be circled.) These roles are reversed for the momentum cells.

The sketches at the bottom of Figure 2 help to show how the momentum cell is drawn. It appears that one takes a junction, such as 3, and adds together about one half of the volumes 2 and 3 (circled) in each side of it. In this way a piece, such as the top of the lower plenum, forms part of more than one momentum cell, as in the central and right-hand figures. The bottom part of the lower plenum apparently forms part of nothing and might as well not be there as far as the momentum balances go. It is difficult to relate these cells to the "generalized control volumes" on Page II-82 as that would seem to make the flow come out of the bottom of the volumes in

Figure 2 and go into the bottom of the lower plenum with no way to get out. The specific examples do not seem compatible with the "generalized" approach.

Tables 1 and 2 are baffling, apart from the directions associated with the arrows drawn at junctions which appear in the second column in Table 1. Because the momentum cells are staggered from the others, the momentum flux terms at the boundaries of a momentum cell do not correspond to these "junctions" but should be evaluated at the boundaries of the shaded volumes in the lower figures, where the Ws have bars and the "junctions" have circles. The Tables appear to contain a mixture of what appear to be Ws to be used to evaluate flux terms for the uncircled junctions and Ws to be used to describe the average momentum in the circled ones. The text below Table 1 states "The momentum flux terms are evaluated using the averaging model for the volume centered flows, where the volume centered flow is the arithmetic average of the inlet and exit flows." There is no explanation of how averaging led to the entries in Tables 1 or 2. "The actual equations implemented in RETRAN-3D to perform this task are given in Appendix A," but it is no help because it is not explained how the general equations are applied to the particular example.

It would be very desirable to have the actual momentum equations deduced from these tables presented in full. It should also be made clear what the specific values of all the terms actually are and how they are evaluated. This would help to clarify the procedures to be applied by a user and to remove ambiguities that remain in the present definitions and methods. It would additionally make it possible to evaluate the reasonableness of the results, as was done above for the bend and tee-junction.

Describing what appear to be some of the ambiguities and uncertainties with the existing documentation may help EPRI to respond more fully. For example, the W_k and W_{k+1} terms with bars are defined to be the flow rates into and out of a momentum cell. They seem to be resolved into a direction ψ, though scalars cannot be resolved. In Table 1, it seems that at Junction 2 W_2 goes in and $1/2W_2$ comes out. This does not correspond to any identifiable cell in Figure 2. One-half of W_2 is not the flow into or out of any region. One-half is not the cosine of any angle of relevance to the situation even if flows could be resolved. In the next line, Junction 3 has $1/2\ W_3$ going in and nothing coming out. This is probably another example of the "interpolation" that gave strange results that defied the concept of continuity in the bend example.

In any case, there is no indication of how these values might be incorporated into the momentum equation for the shaded region called "Junction 2, cold leg to downcomer" in Figure 2. There is also no discussion of how to evaluate the "L" factor in the transient term and what appropriate "W" to use there. Therefore, this example does little to help the user understand the approach.

The text on Page 10 does not help either. If steady state conditions are assumed so that "$W_1=W_2=W_3$" then how is this compatible with a "transient" analysis? Why is W_1 with a bar "simply W_2" and not something like $(W_1+W_2)/2$? Flow rates do not have components so how can x- and y- components be defined, and how can they be deduced to be 1/2 W_2 which seems physically unreasonable?

The average orientation of the shaded volume is called theta and said to be 315 degrees (not a volume average) but this is not the same as ψ and anyway there is no theta in equation (5) so it is unclear what this is to be used for. At the end of the discussion of Junction 2 on Page 10 it is said that the factor 1/4 arises because of angular effects. Now, remember that the TP+J model is a scalar model (see the "Discussion") and the ρv^2 terms do not have to be "resolved" any more than the pressure terms do, so there are really no "angular effects" if this model is being used. During the meeting on February 20, a few examples were given to show how these hypothesized "angular effects" could give rise to significantly different results, for example at the tee-junction between the surge line and the hot leg, that might influence flow distribution during a transient.

Looking briefly at the other examples involving the lower plenum, it is unclear why the Junctions 5 and 6 are said to have no momentum flux when they have flows through them, why W_5 plays no role, and how W_4 can describe the momentum in the sum of the two shaded partial volumes for volume 4. In Table 2, it looks as if Volume 3, presumably the momentum cell around Junction 3, has no momentum in it; why? What pressure terms are to be used to describe Junctions 3 and 4? They have four boundaries that connect to regions containing other fluid. Equation (5) only has two pressures in it.

In reality, the lower plenum part of the reactor vessel is like a turbine bucket that turns the flow coming down out of the downcomer around in the direction of the core. A momentum balance would have to include the force from this structure. If, on the other hand, this is to be modeled as a TP+J, so that Equation (9) in the "Discussion" can be used to describe it, then it is unclear how the shaded volumes as drawn can be forced into such a conceptual framework. The various sketches of "general" volumes, such as Figures II.3-5 and II.3-6, do not help explain either the basis of the general RETRAN equation or how it is used to analyze a case like this.

References:

1. Letter dated October 22, 1999, from G. B. Swindlehurst, Duke Power Company, to Document Control Desk, NRC, Subject: Project No. 669 - Review of RETRAN-3D Response to RAI Letter Dated August 25, 1999.
2. NP-7450, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI, October 1996.
3. G. F. Niederauer, C. E. Peterson, E. D. Hughes, W. G. Choe, "Application of RETRAN to Complex Geometries: Two-Dimensional Hydraulic Calculations," EPRI NP-1415, 1980.
4. T. A. Porsching, "A Scalar Macroscopic Momentum Balance for Multi-dimensional Fluid Flow," October 15, 1999.
5. R. B. Bird, W. E. Stewart, and E. N. Lightfoot, "Transport Phenomena, John Wiley & Sons, New York, NY, 1976.
6. T. A. Porsching, "Scalar Macroscopic Momentum Balances for Multi-dimensional Fluid Flow," April 18, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED FINAL REGULATORY GUIDE, DG-1069, "FIRE PROTECTION PROGRAM FOR NUCLEAR POWER PLANTS DURING DECOMMISSIONING AND PERMANENT SHUTDOWN," DATED FEBRUARY 26, 2001

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, the Committee considered the subject proposed final regulatory guide and decided not to review it. The Committee has no objection to issuing the final regulatory guide for industry use.

Reference:
Memorandum to Dana A. Powers, Chairman, ACRS, B. John Garrick, Chairman, Advisory Committee on Nuclear Waste, Joseph A. Murphy, Chairman, Committee to Review Generic Requirements, from Roy P. Zimmerman, Deputy Director, Office of Nuclear Reactor Regulation, Subject: DG-1069, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," dated February 26, 2001.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
S. Collins, NRR
G. Holahan, NRR
E. Connell, NRR
J. Hickman, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED AMENDMENT TO 10 CFR 50.55a, "CODES AND STANDARDS"

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, the Committee considered the proposed amendment to 10 CFR 50.55a, "Codes and Standards," and decided not to review it. The Committee has no objection to issuing the proposed amendment for public comment.

Reference:
Memorandum dated April 3, 2001, from Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, to William D. Travers, Executive Director for Operations, Subject: Proposed Amendment to 10 CFR 50.55a, "Codes and Standards."

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
J. Strosnider, NRR
E. Imbro, NRR
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT COMMISSION PAPER REGARDING THE SAFEGUARDS PERFORMANCE ASSESSMENT PILOT PROGRAM

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, the Committee considered the proposed Safeguards Performance Assessment (SPA) Pilot Program and decided not to review it. The Committee has no objection to the staff's plan to implement the SPA Pilot program subsequent to Commission approval.

Reference:
Memorandum dated April 3, 2001, from Glenn Tracy, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, transmitting draft copy of Commission paper regarding Safeguards Performance Assessment Pilot Program.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
B. Boger, NRR
G. Tracy, NRR
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 13, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED FINAL LICENSE RENEWAL GUIDANCE DOCUMENTS

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, we reviewed the proposed final versions of NUREG–1800, "Standard Review Plan for Review of License Renewal Applications;" NUREG-1801, "Generic Aging Lessons Learned (GALL) Report;" Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses;" and NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule." These documents provide guidance for preparing and reviewing license renewal applications. Our Subcommittee on Plant License Renewal met on March 27, 2001, to review these documents. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The license renewal guidance documents should be approved for issuance.

2. The staff should encourage applicants to include the results of the scoping process in their applications. The availability of these results will facilitate the review process significantly and make license renewal applications more understandable.

3. The staff has agreed to update the GALL report periodically. The staff should also update the Standard Review Plan (SRP) and Regulatory Guide 1.188 to make them consistent with the updated GALL report.

Discussion

We reviewed earlier drafts of the license renewal guidance documents during our November 2-4, 2000 meeting and provided comments and recommendations in a report dated November 15, 2000. We concluded at that time that the draft documents described a consistent and understandable process to support the preparation and review of license renewal applications.

The current versions of the guidance documents include the resolution of comments provided by industry, the ACRS, and the public. The documents were revised to increase focus, eliminate insignificant aging effects, improve clarity, and to include changes resulting from the resolution of technical issues between the staff and the industry. The documents now provide closure for the great majority of aging management issues. The staff is continuing its dialogue with NEI and current applicants on a number of residual issues. This dialogue is likely to continue for some time as issues are closed and other issues are identified as a result of the lessons learned from reviewing future license renewal applications.

The development and staff review of previous license renewal applications would have been facilitated by the availability of a clearly defined baseline for regulatory acceptance that the guidance documents now provide. Given the significant number of applications being developed by licensees and reviewed by the staff, we agree with the staff and NEI that these documents should be approved. Changes resulting from the continuing dialogue between the staff and the industry can be incorporated into future updates. The staff has agreed to update the GALL report periodically. It should also update the SRP and Regulatory Guide 1.188 accordingly.

The License Renewal rule requires that for those structures, systems, and components (SSCs) that are within the scope of license renewal, applicants identify structures and components that are subject to an aging management review (AMR). The industry has taken the position that an applicant needs to include in its application only a description of the methodology used to implement the scoping and screening processes and the results of the screening process (i.e., the list of structures and components identified as requiring an AMR). The staff has modified the guidance documents to reflect this position. This meets the requirement of the rule, but our experience with past license renewal applications is that the scoping process is complex, and the lack of the scoping process results in the application (i.e., the list of SSCs that are within the scope of license renewal) represents a significant challenge for the reviewers and interested members of the public. An application that includes the results of the scoping process is more scrutable, facilitates the staff's review, and assists the staff in determining that structures and components subject to an AMR have been consistently identified. Without this information, the staff must rely on requests for additional information, site audits, and limited sampling of components not selected for an AMR. Inclusion of this information in the application does not constitute a significant added burden to the applicant and should, therefore, be encouraged.

Sincerely,

George E. Apostolakis
Chairman

<u>References</u>

1. U.S. Nuclear Regulatory Commission, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated March 1, 2001.
2. U.S. Nuclear Regulatory Commission, NUREG-1801, Vol. 1, "Generic Aging Lessons Learned (GALL) Report," dated March 1, 2001.
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," March 2001.
4. NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR 54 - The License Renewal Rule," March 2001.
5. U. S. Nuclear Regulatory Commission, NUREG-1739, "Analysis of Public Comments on the Improved License Renewal Guidance Documents," dated March 1, 2001.
6. Report dated November 15, 2000, from D. A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to R. A. Meserve, Chairman, U.S. Nuclear Regulatory Commission, Subject: License Renewal Guidance Documents.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 13, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C., 20555-0001

Dear Dr. Travers:

SUBJECT: CLOSURE OF GENERIC SAFETY ISSUE-170, "REACTIVITY TRANSIENTS AND FUEL DAMAGE CRITERIA FOR HIGH BURNUP FUEL"

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, we discussed the proposed closure of Generic Safety Issue-170 (GSI-170), "Reactivity Transients and Fuel Damage Criteria for High Burnup Fuel." We also had the benefit of the documents referenced.

GSI-170 was originally initiated in 1995 to address the effects of high fuel burnup on fuel damage limits during design basis accidents such as reactivity transients. The staff has imposed limits on fuel burnup because there are not adequate data on fuel behavior at higher levels of burnup.

The staff has undertaken research programs to confirm the regulatory decision on the allowable level of fuel burnup. In our letter dated March 24, 1999, we stated that conducting an expert opinion elicitation to identify and rank important phenomena that affect high burnup fuel would provide a sound technical basis for refining the staff's confirmatory research program. We also stated that the expert opinion elicitation could provide technical bases for establishing the data and analyses needed to support applications for extending fuel burnup beyond current regulatory limits.

Since then, the staff has held a series of meetings with expert panels and carried out phenomena identification and ranking elicitations for loss-of-coolant accidents in pressurized water reactors (PWRs) and boiling water reactors (BWRs), rod ejection accidents in PWRs, and power oscillations in BWRs. The technical issues that are listed in GSI-170 have been clarified, and well-defined research programs are being pursued to address these issues.

We have, therefore, no objection to the proposed closure of GSI-170.

Sincerely,

George E. Apostolakis
Chairman

References:
1. Memorandum dated February 6, 2001, from Farouk Eltawila, Acting Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, Subject: Closure of Generic Issue-170, Reactivity Transients and Fuel Damage Criteria for High Burnup Fuel.
2. Letter dated March 24, 1999, from Dana A. Powers, Chairman, ACRS, to William. D. Travers, Executive Director for Operations, Subject: High Burnup Fuel Phenomena Identification and Ranking.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 16, 2001

Dr. William D. Travers
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: INTERIM LETTER RELATED TO THE LICENSE RENEWAL OF EDWIN I. HATCH NUCLEAR STATION, UNITS 1 AND 2

During the 481st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2001, we reviewed the NRC staff's Safety Evaluation Report (SER) Related to the License Renewal of Edwin I. Hatch Nuclear Station, Units 1 and 2. Our Subcommittee on Plant License Renewal also reviewed this matter on March 28, 2001. During our review, we had the benefit of discussions with representatives of the NRC staff and the Southern Nuclear Operating Company, Inc. (SNC), and of the documents referenced.

Conclusions

1. The staff performed an extensive and thorough review of the license renewal application for Hatch, Units 1 and 2. Although a number of open issues are yet to be resolved, the staff has concluded that SNC has implemented adequate processes to identify structures, systems, and components (SSCs) subject to an aging management review and to manage age-induced degradation of these SSCs. We concur with the staff.

2. SNC incorporated by reference several Boiling Water Reactor Vessel and Internals Project (BWRVIP) topical reports into the Hatch license renewal application. We agree with the staff that the guidelines in the BWRVIP topical reports effectively support license renewal.

Discussion

By letter dated February 29, 2000, SNC submitted the license renewal application for Hatch, Units 1 and 2, in accordance with 10 CFR Part 54. SNC requested renewal of the operating licenses for the Hatch units for a period of 20 years beyond the current license expiration dates of August 6, 2014, for Unit 1 and June 13, 2018, for Unit 2.

The SER documents the results of the staff's review of the Hatch license renewal application and additional information submitted by SNC through January 31, 2001. The staff's review included verification of the completeness of the identification of the SSCs within the scope of the License Renewal rule, the validation of the plant assessment process, the identification of the possible aging mechanisms associated with each passive long-lived component, and the adequacy of the aging management programs. The staff also conducted onsite inspections to verify the adequacy of the implementation of the programs described in the application. The staff's review of the license renewal application for Hatch was extensive and thorough.

The SNC approach to the identification of SSCs that are within the scope of the rule is function based rather than system based as was the case in previous applications. This approach led to correct identification of SSCs within the scope of the rule. However, as implemented, this approach made it difficult for the reviewers to ascertain which SSCs were in scope and which were not. This experience emphasizes the importance of a proper choice of scoping and screening processes in facilitating the review process and in making the application more scrutable, especially to interested members of the public.

To confirm the adequacy of the methodology, the staff had to rely heavily on the review of supporting documents located at the site and on requests for additional information. The staff also performed a "walkthrough" of the process for three systems at Hatch. This review was thorough, provided adequate evidence that SNC had identified SSCs in scope, and identified improvements in supporting procedures to enhance the repeatability of the scoping and screening processes.

The BWRVIP has developed topical reports that provide guidelines for inspection, evaluation, repair, and mitigation of aging degradation of vessels and the internals in BWRs. This program was expanded to include explicit consideration of provisions for license renewal. This extensive program is documented in over 20 topical reports. The staff has reviewed and approved most of these reports. Approval of the remaining reports awaits closure of related open items.

Hatch has used the guidance provided in the BWRVIP topical reports in the development of many of its aging management programs. Indications of cracking in several reactor vessel internal components identified at Hatch have been dispositioned either by repair or by monitoring according to BWRVIP guidelines. The large number of BWR licensees committed to the BWRVIP program provide a continuous flow of new inspection and evaluation data that either confirm the adequacy of the programmatic initiatives or will provide an early warning system should unexpected degradation occur.

We reviewed BWRVIP topical reports 26, 41, and 75 that address the top guide, the jet pump assembly, and inspection procedures and schedules for piping. We concur with the staff that these topical reports provide an acceptable demonstration that the effects of aging on these components can be adequately managed during the period of extended operation.

Sincerely,

George E. Apostolakis

George E. Apostolakis
Chairman

References:
1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report With Open Items Related to the License Renewal of Edwin I. Hatch, Units 1 and 2," February 2001.
2. Letter from H. L. Sumner to the U. S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Application for Renewed Operating Licenses," dated February 29, 2000.
3. Topical Report BWRVIP-26, "Top Guide - Inspection and Flaw Evaluation Guidelines," dated December 27, 1996.
4. Topical Report BWRVIP-41, "BWR Jet Pump Assembly - Inspection and Flaw Evaluation Guidelines," dated October 15, 1997.
5. Topical Report BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313)," dated October 27, 1999.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 15, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: for John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED FINAL RULEMAKING TO AMEND 10 CFR PART 55 AND ASSOCIATED REGULATORY GUIDE 1.149, REVISION 3

During the 482nd meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 2001, the Committee considered the proposed final rulemaking to amend 10 CFR Part 55, "Operators' Licenses," regarding operator license eligibility and the use of simulation facilities in operator licensing; and proposed final Revision 3 of Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations," and decided not to review these documents. The Committee has no objection to issuing the proposed final rule and associated Regulatory Guide 1.149, Revision 3.

Reference:
Memorandum dated May 2, 2001, from Bruce A. Boger, Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Final Rulemaking to Amend 10 CFR Part 55, "Operators' Licenses," Regarding Operator License Eligibility and the Use of Simulation Facilities in Operator Licensing; and Final Revision 3 of Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations."

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
B. Boger, NRR
C. Goodman, NRR
L. Vick, NRR
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 18, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR ARKANSAS NUCLEAR ONE, UNIT 1

During the 482nd meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 2001, we completed our review of Entergy Operations, Inc., application for license renewal of Arkansas Nuclear One, Unit 1 (ANO-1), and the related final Safety Evaluation Report (SER). Our review included two meetings with the staff and the applicant. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. Entergy has properly identified the structures, systems, and components (SSCs) that are subject to aging management review consistent with the requirements of 10 CFR Part 54.

2. Aging mechanisms associated with passive, long-lived SSCs have been appropriately identified.

3. The programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that ANO-1 can be operated in accordance with its current licensing basis for the extended license term without undue risk to the health and safety of the public. The programs do not explicitly address the potential for circumferential cracking in control rod drive mechanism (CRDM) nozzle penetrations, such as has been observed at the Oconee Nuclear Plant, Unit 3. We expect that this current problem will be resolved and that the resolution will be incorporated into the current licensing basis and carried over into the license renewal period.

4. The staff has performed a comprehensive and thorough review of Entergy's application, and the open items identified in the January 2001 draft SER have been satisfactorily resolved .

5. The staff should determine whether modification of the current guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is required to reflect the lessons learned from the ANO-1 application regarding aging management of small-bore piping and medium-voltage buried cable.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. Entergy requested renewal of the operating license for ANO-1 for a period of 20 years beyond the current license term, which expires on May 20, 2014. The final SER documents the results of the staff's review of information submitted by Entergy, including those commitments that were necessary to resolve open items identified by the staff in its January 2001 draft SER. The staff's review included verification of the completeness of the SSCs identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging mechanisms associated with each passive long-lived component, and the adequacy of the aging management programs.

Our Subcommittee on Plant License Renewal met with the applicant and the staff on February 22, 2001, to review the SER with open items. The Subcommittee did not identify any issues to be addressed other than the six open items identified by the staff. This remarkably small number of open items is due, in large part, to the fact that the applicant implemented relevant lessons learned from the previous license renewal applications. In addition, the applicant structured the application using the standard application format and the guidance in Nuclear Energy Institute (NEI) Report 95-10, which facilitated the review. Because of the small number of open items and the scrutability of the application, we decided that there was no necessity to provide an interim report and have reviewed the SER on an accelerated basis.

The process implemented by the applicant to identify SSCs within the scope of the License Renewal Rule is effective. Reactor coolant system (RCS) components were identified using the generic Babcock & Wilcox Owners Group (BWOG) topical reports that address aging of RCS piping, pressurizer, reactor vessel, and reactor vessel internals. These topical reports, which have been approved by the staff, are applicable to ANO-1 and were used to support the license renewal application for Oconee. All other components in scope were determined on a plant-specific basis. At ANO-1, the safety-related SSCs included in the quality assurance program ("Q" list), as required by 10 CFR Part

50, Appendix B, are those that meet the definition of "safety related" in 10 CFR 54.4(a)(1). Furthermore, the majority of SSCs whose failure could prevent satisfactory accomplishment of any of the safety-related functions in 10 CFR 54.4(a)(1) are also classified as safety-related and included in the ANO-1 "Q" list. Therefore, the applicant was able to use the "Q" list to identify the bulk of the ANO-1 SSCs within the scope of the License Renewal Rule. This process has also resulted in the conservative inclusion of some SSCs that do not meet the criteria of 10 CFR 54.4(a)(2). We concur with the staff that the applicant has properly identified SSCs requiring an aging management review.

The applicant conducted a comprehensive aging management review of SSCs in scope. Aging effects of RCS components were identified using the aforementioned BWOG topical reports. Aging effects of all other SSCs were identified based on component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Appendix B of the application describes the 22 existing or modified programs and the seven new programs implemented to manage aging during the period of extended operation.

ANO-1 has proposed a significantly smaller number of one-time inspections than did previous applicants. This is due, in part, to the fact that existing or modified ANO-1 programs manage aging effects that previous applicants do not manage during their current license terms. Consequently, previous applicants had to implement a larger number of one-time inspections to support license renewal. For example, aging of small-bore piping is managed at ANO-1 by a plant-specific risk-informed inspection program, and therefore, does not require a one-time inspection. We agree with the staff that the applicant has properly identified possible aging mechanisms associated with passive, long-lived SSCs and that the programs instituted to manage aging degradation of the identified SSCs are appropriate.

The ANO-1 application identifies cracking at welded joints of the CRDM pressure boundary as an aging effect to be managed. Appendix B of the application describes the aging management program instituted to deal with this aging degradation mechanism; i.e., "CRDM nozzle and other vessel closure penetration inspection program." This program identifies primary water stress corrosion cracking of Alloy-600 nozzles with partial penetration welds as the aging effect of concern and ties programmatic elements, such as the frequency of inspections, to the results of plant-specific and sister plant inspection findings. The initiatives included in this program are adequate to deal with this identified aging effect during the remaining portion of the current license term and during the period of extended operation. However, it is likely that the recent observations of stress corrosion cracking at the outer surface of CRDM nozzle penetrations may require some revisions to the program. We have noted previously that aging management programs may have to be revised if it is found that new modes of degradation are occurring.

The ANO-1 application includes time limited aging analyses (TLAA) to evaluate the impact of neutron embrittlement on reactor vessel integrity. These analyses determine reactor vessel resistance to failure during pressurized thermal shock (PTS) events and the maintenance of acceptable Charpy upper-shelf energy levels. The TLAA used the methodology described in topical report BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel." This topical report was reviewed and approved by the staff and reviewed by the ACRS. Based on the composition of the limiting welds, Entergy projected that the ANO-1 reactor vessel will not reach the PTS and Charpy upper-shelf energy screening limits until well after 60 years of operation. The ANO-1 reactor vessel integrity program will be utilized to ensure that the time-dependent parameters used in the TLAA evaluations are tracked so that the TLAA remain valid during the license renewal period.

Entergy committed to implementing a plant-specific program to manage the effects of fatigue. Using the correlations published in NUREG/CR-5704, Entergy has found that the surge line, the high pressure injection/makeup nozzles, and safe ends may reach the limits of acceptable fatigue during the period of extended operation. To address this condition, Entergy has proposed a program that will include one or more of the following options: refinement of the fatigue analyses, repair, replacement, or management of fatigue effects using a program that will be reviewed and approved by the staff. We concur with the staff that Entergy's proposed program is an acceptable plant-specific approach for resolving the concerns of Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60 Year Plant Life."

ANO-1 region 1 spent fuel storage racks currently use Boraflex as a neutron absorber. Aging of Boraflex was identified in the application as a time limited aging analysis. During the staff's review of the ANO-1 application, Entergy informed the staff that Boraflex had been found to degrade more rapidly than previously expected, and was not expected to last through the current 40-year licensing term. Therefore, a corrective action plan for the remainder of the 60-year operating term would be identified and committed to before the end of 2002. In Open Item 4.7.2-1 associated with Boraflex degradation, the staff requested that Entergy continue to recognize aging of Boraflex as a time limited aging analysis and provide details on the required monitoring program. Entergy has now provided the requested programmatic details. We concur with the staff that either the implementation of a permanent solution during the current licensing period or the Boraflex monitoring program provided by Entergy and described in the SER provides acceptable management of Boraflex degradation during the period of extended operation.

The staff has performed a comprehensive and thorough review of Entergy's application. The applicant and the staff have identified possible aging mechanisms associated with passive long-lived components. Adequate programs have been established to manage the effects of aging so that ANO-1 can be operated safely in accordance with its current licensing basis for the extended license term.

The review of the ANO-1 application has provided significant new information on small-bore piping and medium-voltage buried cable aging degradation and related management programs. As described above, ANO-1 has implemented a small-bore piping inspection program because it has identified small-bore piping in safety-significant locations that is susceptible to aging degradation. The staff should determine whether current guidance in the GALL report needs to be modified to reflect this experience. Also, ANO-1 has implemented a medium-voltage buried cable aging management program that includes the options of cable testing or periodic replacement of buried cables. ANO-1 has included the replacement option because it has found that in a number of instances testing was not effective in identifying cable degradation. The staff needs to evaluate the adequacy of testing of buried cables and provide appropriate guidance in the next update of the GALL report.

Dr. William J. Shack did not participate in the Committee's deliberations on aging-induced degradation.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1," dated April 2001.
2. Letter dated January 31, 2000, from C. R. Hutchinson to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Application.
3. Letter dated March 14, 2001, from J. D. Vandergrift to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Safety Evaluation Report Open Item Responses.

4. Babcock and Wilcox Owners Group Generic License Renewal Program Topical Report, BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," dated June 1996.
5. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Steels," dated April 1999.
6. U. S. Nuclear Regulatory Commission, Generic Safety Issue - 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 18, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL MANAGEMENT DIRECTIVE 6.4, "GENERIC ISSUE PROGRAM"

During the 482ND meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 2001, we completed our review of the proposed final Management Directive 6.4, "Generic Issue Program." During our 480th meeting, March 1-3, 2001, we discussed this matter with representatives of the NRC staff. We had the benefit of the documents referenced.

Conclusion

The proposed Management Directive 6.4 and the associated handbook, modified as appropriate based on the results of the pilot study, should provide an effective way to implement the revised generic issue process.

Discussion

The Office of Nuclear Regulatory Research (RES) has established criteria and guidance for risk-informed technical screening of generic issues applicable to reactor and materials facilities. The technical screening is to evaluate the possible safety implications of generic issues in a disciplined, quantitative manner. This approach, which uses probabilistic risk assessment, is comprehensive and provides an improved basis for decisionmaking. The screening of generic issues uses risk insights related to changes in core damage frequency, large early release frequency, and the product of the frequency of an accident and the averted public dose (person-rems). We agree with this approach, particularly since all three of the above risk metrics are to be used in the decisionmaking process.

We reviewed the reevaluation of the generic issue process along with the proposed Management Directive 6.4, and the associated handbook to implement the revised generic issue process and made a number of recommendations in our letter of April 19, 1999. One of these recommendations was that the staff conduct a pilot study to evaluate the effectiveness of using the Management Directive for implementing the revised generic issue process.

The RES staff conducted a pilot study. The results were very informative and the staff gained significant insights related to implementation problems. As a result, the staff developed a number of recommendations on how to improve the generic issue process and its implementation. We agree with these recommendations.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated April 11, 2001, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to Directors of NRC Offices, Subject: Management Directive 6.4, "Generic Issue Program."
2. Letter dated April 19, 1999, from Dana A. Powers, ACRS Chairman, to William D. Travers, Executive Director for Operations, Subject: Reevaluation of Generic Safety Issue Process.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 11, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED FINAL REGULATORY GUIDE, 1.52, REVISION 3, "DESIGN, INSPECTION, AND TESTING CRITERIA FOR AIR FILTRATION AND ADSORPTION UNITS OF POST-ACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEMS IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS,"

During the 483rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2001, the Committee considered the subject proposed final regulatory guide and decided not to review it. The Committee has no objection to issuing the final regulatory guide for industry use.

Reference:
Memorandum to ACRS Members , from P. Boehnert, ACRS Staff, Subject: Revision 3 to Regulatory Guide 1.52 (DG-1102): Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident-Engineered-Safety-Feature (ESF) Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants, dated May 24, 2001.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
S. Collins, NRR
B. Sheron, NRR
G. Holahan, NRR
J. Hannon, NRR
J. Segala, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 12, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED REVISION 1 TO RISK-INFORMED REGULATORY GUIDE 1.174 AND STANDARD REVIEW PLAN CHAPTER 19

During the 483rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2001, the Committee considered the proposed Revision 1 to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and associated revision to NUREG-800, Standard Review Plan (SRP), Chapter 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific Risk-Informed Decisionmaking: General Guidance." The Committee plans to review the proposed final version of Regulatory Guide 1.174 and SRP Chapter 19.0 following the reconciliation of public comments. The Committee has no objection to issuing these documents for public comment.

Reference:
Memorandum dated June 1, 2001, from Thomas L. King, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Revision 1 to Risk-Informed Regulatory Guide 1.174 Standard Review Plan Chapter 19.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
T. King, RES
M. Cunningham, RES
S. Collins, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 14, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

SUBJECT: RESPONSE TO YOUR MAY 7, 2001 MEMORANDUM REGARDING DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE ISSUES

Dear Chairman Meserve:

This report responds to the May 7, 2001 memorandum in which you requested our views on whether immediate actions are needed, other than those already taken by the staff, to deal with steam generator tube issues. In February 2001, we submitted to the Executive Director for Operations (EDO) NUREG-1740 on a differing professional opinion (DPO) concerning alternative repair criteria for steam generator tubes. In that report, we concluded that alternative repair criteria were needed. The alternative repair criteria and the condition monitoring program for steam generator tubes that the staff has endorsed can provide adequate protection of the public health and safety.

We did make recommendations to the EDO directed particularly at improving the technical bases of the alternative repair criteria and the reliability of the condition monitoring program. The more important of these recommendations are:

- Evaluate the potential for progression of damage to steam generator tubes during rapid depressurization caused by a main steamline rupture.

- Monitor performance to search for systematic deviations from the linear bound on the nonlinear processes of crack initiation and growth through steam generator tube walls.

- Improve the database for the correlation of leakage with voltage for 7/8" tubes.

- Improve the analysis and understanding of radioiodine behavior during design basis accidents.

- Develop a better understanding of the behavior of degraded steam generator tubes under severe accident conditions.

We did not identify issues that demanded immediate, pre-emptory resolution for the alternative repair criteria and the condition monitoring program to continue. We felt that the recommended activities could be done within the context of the existing Action Plan on Steam Generators. Research needed to act upon the recommendations could be prioritized and pursued within the context of the current research program. We did encourage the staff to determine promptly whether the effects of forces associated with depressurization during a main steamline break constitute a generic safety issue and, if so, to resolve this issue expeditiously.

We find the approach the EDO has taken so far in response to our recommendations to be appropriate. We look forward to reviewing the details of the staff's responses to our recommendations.

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

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated May 7, 2001, from Richard A. Meserve, NRC Chairman, to George Apostolakis, ACRS Chairman, Subject: Differing Professional Opinion on Steam Generator Tube Issues.
2. U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1740, "Voltage-Based Alternative Repair Criteria," February 2001.
3. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Action Plan on Steam Generators.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 19, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

SUBJECT: RESPONSE TO YOUR APRIL 12, 2001 LETTER ON ISSUES RAISED BY ACRS PERTAINING TO INDUSTRY USE OF THERMAL-HYDRAULIC CODES

Dear Dr. Travers:

Thank you for your letter of April 12, 2001, in which you describe the actions that the staff is taking to manage the issues raised by the ACRS concerning thermal-hydraulic codes. We suggest that you reconsider two of your responses.

In response to our Recommendation 6, you state that the study would "require substantial resources." This assessment needs to be balanced against the cost to the NRC in credibility with the informed technical community, and perhaps eventually in public safety, of continuing to approve codes that for decades have contained questionable simplifications at a fundamental level. Some of these simplifications are extreme enough to invite serious questions by expert reviewers regarding their adequacy. We suggest that the staff assess a range of appropriate studies to justify these simplifications. In addition to increasing confidence in code predictions, results that could be published in the open literature would help to reassure the technical community that these codes work for good reasons and would support the Commission's Performance Goal of increasing public confidence.

Furthermore, you suggest that the present system, based on PIRT (Phenomena Identification and Ranking Table) is adequate to address the issue. PIRT is a method whereby experts agree on the important phenomena to model in a code. Although in some cases it may provide insight into why a code may give satisfactory results despite limitations, the PIRT does not address the question of how adequately the phenomena are actually modeled.

The "PIRT-based assessment matrix," mentioned in your letter as being used to assess the TRAC-M code of the Office of Nuclear Regulatory Research appears to be an attempt to tie the assessment process more rigorously to the results of the PIRT. We look forward to discussing the results of this improvement with the staff later this year. In the past, the code assessment process has been overly qualitative and has permitted the persistence of weak elements in the codes.

We also wish to clarify Recommendations 8 and 9.

We accept that both the Code Scaling, Applicability, and Uncertainty evaluation methodology and draft Regulatory Guide DG-1096 address the importance of uncertainty analysis. We are also aware that Regulatory Guide 1.174 recognizes that many sources of uncertainty are not readily quantifiable at the present time.

What concerns us is an excess of leeway in the expectations of the staff. This allows treatment of uncertainties to take place in an atmosphere of negotiation in which many arguments are qualitative and criteria for evaluation are unspecified. We believe that the NRC should move toward establishing a mature process for evaluating uncertainties — a process that has an intellectual backbone, is validated by data and experience, and can be clearly communicated to the informed public.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated April 12, 2001, from William D. Travers, Executive Director for Operations, NRC, to George E. Apostolakis, Chairman, ACRS, Subject: Issues Associated With Industry-Developed Thermal-Hydraulic Codes.
2. Report dated January 11, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Issues Associated With Industry-Developed Thermal-Hydraulic Codes.
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide, DG-1096, "Transient and Accident Analysis Methods," dated July 18, 2000.
4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 19, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: RISK-BASED PERFORMANCE INDICATORS: PHASE 1 REPORT

During the 483rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2001, we completed our review of the staff's report on the results of the Phase 1 effort to develop risk-based performance indicators (RBPIs). We also discussed this matter with representatives of the NRC staff and the Nuclear Energy Institute (NEI) during our 482nd meeting, May 10-11, 2001. Our Subcommittee on Reliability and Probabilistic Risk Assessment discussed this matter on April 17, 2001. We had the benefit of the documents referenced.

Conclusions and Recommendations

Although the scope of the Phase 1 study was limited to the technical feasibility of developing RBPIs, our letter addresses some of the questions that might arise in the implementation of these candidate PIs.

1. A rational framework has been established for evaluating RBPIs and handling the relevant aleatory and epistemic uncertainties in evaluating PIs from available data.

2. The staff should continue to develop RBPIs as part of the ongoing effort to make the reactor oversight process (ROP) more objective and scrutable.

3. The staff should develop methods for assessing tradeoffs between introducing new PIs versus reducing baseline inspections.

4. The staff should investigate establishing thresholds that depend on the baseline core damage frequency (CDF) of the plant.

5. The Phase 1 report states that the green/white thresholds used in the current ROP correspond to changes in CDF (ΔCDF) that vary by more than an order of magnitude among plants. The green/white thresholds in the ROP should be reevaluated.

6. The derivations of decision rules (thresholds for RBPIs) given in Appendix F to the RBPI Phase 1 report should be expanded to include plant- or design-specific prior distributions.

7. The staff should continue to explore "alternative" RBPIs.

8. The potential for unintended impacts of RBPIs on plant performance is a concern and should be carefully considered in the development of the RBPIs.

9. The staff does not have the up-to-date risk information needed to develop RBPIs for shutdown operations; therefore, the staff's work should focus on full-power operations until such information is developed.

10. There should be a publicly available peer review of the SAPHIRE code and, eventually, the Standardized Plant Analysis Risk (SPAR) models.

11. It is premature to initiate a pilot program for RPBIs.

Discussion

PIs and baseline inspections constitute major elements of the ROP whose objective is to verify that reactor facilities are operated safely and to provide early warning of adverse trends and deteriorating licensee performance. The PI values are determined based on statistical evidence from actual plant performance and, therefore, remove some of the subjectivity that is inherent in the inspection and assessment processes. Even though inspection findings are related to risk metrics through the significance determination process (SDP), the PIs are less subjective. Both PIs and inspection findings provide input to the action matrix in determining the need for increased NRC involvement in addressing plant performance issues. RBPIs have the advantage that their relation to risk is direct and more transparent than that of other types of PIs.

Although the evaluation of PIs appears to be a straightforward and objective process, it is important to distinguish between aleatory and epistemic uncertainties to ensure that the calculated values are statistically meaningful. The practical questions in the evaluation of PIs are: How long should the observation period be and how many occurrences over this period (the aleatory variable) will lead to the conclusion that the average frequency (the epistemic variable) has shifted? Statistical methods for handling these questions are available and have been employed appropriately by the staff in Appendix F. We encourage the staff to continue this work.

It is important that PIs and inspections complement each other and that the collection of redundant information be avoided. Introducing additional PIs should be justified either on the basis that some important aspects of plant performance are not addressed well in the current ROP or that information previously collected via inspections can now be better obtained through the new PIs, thereby allowing reduced inspection. In order to evaluate the potential for increased regulatory burden associated with additional PIs, the staff should develop methods for evaluating the tradeoffs between introducing new PIs and reducing inspections.

In the Phase 1 study, the thresholds between green and white (GW), white and yellow (WY), and yellow and red (YR) performance bands are chosen to correspond to ΔCDF. The GW threshold corresponds to a ΔCDF of 10^{-6}/reactor-year. The ΔCDF values for the WY and YR thresholds are10^{-5}/reactor-year and 10^{-4}/reactor-year, respectively. These values are claimed to be consistent with the acceptance guidelines in Regulatory Guide 1.174. We note, however, that in Regulatory Guide 1.174 the acceptance limits on ΔCDF are 10^{-5}/reactor-year when the baseline CDF is smaller than 10^{-4}/reactor-year and10^{-6}/reactor-year when the baseline CDF is greater than 10^{-4}/reactor-year. The staff should investigate establishing thresholds that correspond to ΔCDF values that are functions of the baseline CDF.

In contrast to the Phase 1 report, the GW thresholds in the ROP are defined in terms of the 95th percentiles of the plant-to-plant variability distributions for a specified reference period. It is noted in the report that, due to the large plant-to-plant variability in the importance of systems, the thresholds in the ROP correspond to ΔCDFs in excess of 10^{-5}/reactor-year for some plants, a value that is an order of magnitude greater than the GW threshold used in the SDP. The choice of the GW threshold in the ROP should be revised so that the ΔCDFs are consistent from plant to plant.

The statistical analyses in Appendix F provide very useful insights into a number of decision rules for determining the thresholds. Appendix F shows that using generic industry information for the occurrence of transients as the prior distribution leads to unrealistic results. For example, for the transient "loss of heat sink," the number of events in a three-year period that must occur to exceed the thresholds are: GW = 19.5, WY = 335.2, and YR = 3,461. Furthermore, regarding component unavailability, it is concluded in Appendix F that only site-specific data are appropriate for estimating the variability of unavailability data at a plant. From this, it is evident that industry-wide prior distributions should not be used.

In addition, two sets of "noninformative" prior distributions are considered in Appendix F. Using these distributions means that, before collecting the data, the analysts assume that they have no knowledge regarding the distribution of the RBPI values. This is too strong an assumption and inconsistent with the information provided in Individual Plant Examinations. As the report states, the RBPIs should reflect *changes* in licensee performance that are logically related to risk. To evaluate these changes, one must start with the existing distributions of the RBPIs (i.e., use the anticipated plant performance as prior distributions) and then incorporate the collected data to determine whether undesired changes have occurred.

To date, PIs have been defined individually. In other words, the thresholds have been set in such a way that, when exceeded, the PI alone indicates unacceptable performance. It is possible, however, that several PIs may increase in such a way that the change in CDF is significant even though each PI remains below its corresponding threshold. The questions are, then: What is an appropriate set of PIs and by how much should they deviate from their expected values to suspect that the licensee performance is indeed deteriorating and that increased regulatory attention is warranted? We commend the staff for raising this very important issue and encourage the staff to pursue what it denotes as "alternative approaches for RBPI determination."

For shutdown modes, the staff is proposing to consider four risk-significant states, depending on reactor-cooling conditions, time after shutdown, and the availability of mitigating system trains. The RBPIs are, then, defined as the times spent in each of these states. NEI raises an important issue regarding an unintended impact of these shutdown RBPIs: thresholds based on time spent in each state could discourage licensees from exercising caution when warranted. For example, a situation may arise while in the "medium" risk state that would call for a deliberate approach, resulting in a longer time in this state. Knowing that this extension of time may move a performance indicator to the white performance band may have an adverse impact on the licensee's decision. There is a similar problem with the unavailability RBPI, which is also based on time (the duration of planned and unplanned outages). These and other unintended impacts should be investigated.

A more fundamental problem with the development of shutdown RBPIs is the lack of adequate risk information. The Phase 1 report had to rely on available results that were based on assumptions that could not be evaluated. The PRA knowledge base for shutdown modes is much weaker than for power operations. In light of this observation and the NEI concern noted above, the staff's work should focus on the development of RBPIs for power operations until sufficient risk information is developed for shutdown modes.

The development of RBPIs uses computerized SPAR models. At this time, about 30 such models have been developed and reviewed by the licensees. We believe that the underlying computer code (SAPHIRE) should be subjected to the Office of Nuclear Regulatory Research process for reviewing computer codes that has been used for SCDAP, CONTAIN, MELCOR, and VICTORIA. Peer review of the SAPHIRE code is a necessary first step that should lead eventually to peer review of the SPAR models.

The Phase 1 report is a good step toward the development of RBPIs. As noted in our recommendations, significant work remains to be done before a pilot program is initiated. We look forward to working with the staff on this important matter in the future.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U. S. Nuclear Regulatory Commission, draft report entitled, Risk-Based Performance Indicators: Results of Phase-1 Development, and Associated Appendices A-G, January 2001.
2. Memorandum dated June 28, 2000, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: SECY-00-0146, "Status of Risk-Based Performance Indicator Development and Related Initiatives."
3. Memorandum dated May 7, 2001, from Hossein G. Hamzehee, Office of Nuclear Regulatory Research, NRC, to Patrick W. Baranowsky, Office of Nuclear Regulatory Research, NRC, Subject: Summary of April 24, 2001 Public Meeting on Draft Phase-1 Risk-Based Performance Indicator Development Report.
4. Memorandum dated March 9, 2001, from Hossein G. Hamzehee, Office of Nuclear Regulatory Research, NRC, to Patrick W. Baranowsky, Office of Nuclear Regulatory

Research, NRC, Subject: Summary of February 21, 2001 Public Meeting on Draft Phase 1 Risk-Based Performance Indicator Development Results.

5. Memorandum dated December 1, 2000, from William M. Dean, Office of Nuclear Reactor Regulation, NRC, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Comments on Draft Phase-1 Risk-Based Performance Indicator Report (Predecisional Draft).
6. Memorandum dated November 30, 2000, from Farouk Eltawila, Office of Nuclear Regulatory Research, NRC, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Review of Report - Results of Phase-1 Risk-Based Performance Indicator Development (Predecisional Draft).
7. Memorandum dated November 20, 2000, from James Wiggins, NRC Region I, to William Dean, Office of Nuclear Reactor Regulation, NRC, Subject: Regional Comments on Report, "Results of Phase-1 Risk-Based Performance Indicator Development" (Predecisional Draft).
8. Memorandum dated November 27, 2000, from Michael E. Mayfield, Office of Nuclear Regulatory Research, NRC, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Review of Report - Results of Phase-1 Risk-Based Performance Indicator Development (Predecisional Draft).
9. Letter dated May 12, 2001, from Stephen D. Floyd, Nuclear Energy Institute, to Michael T. Lesar, Acting Chief, Rules and Directives Branch, NRC, Subject: Comments on "Risk-Based Performance Indicators: Results of Phase-1 Development."
10. Letter dated May 11, 2001, from J. M. Kenny, Chairman, Boiling Water Reactor Owners' Group, to Michael T. Lesar, Division of Administration, NRC, Subject: BWROG Comments on Risk-Based Performance Indicators: Results of Phase-1 Development.
11. Letter dated May 14, 2001, from R. M. Krich, Exelon Generation Company, to Michael T. Lesar, Acting Chief, Rules and Directives Branch, NRC, Subject: Response to Request for Public Comments on Risk-Based Performance Indicators: Results of Phase-1 Development.
12. Letter dated March 9, 2001, from Mark J. Burzynski, Tennessee Valley Authority, to Chief, Rules and Directives Branch, NRC, Subject: Risk-Based Performance Indicators: Results of Phase 1 Development.
13. Memorandum dated January 8, 1999, from William D. Travers, Executive Director for Operations, to the Commissioners, Subject: SECY-99-007, "Recommendations for Reactor Oversight Process Improvements."
14. Memorandum dated March 22, 1999, from William D. Travers, Executive Director for Operations, to the Commissioners, Subject: SECY-99-007A, "Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007)."
15. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 17, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDE DG-1108, "COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE" -- PROPOSED REVISION 2

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, the Committee considered draft Regulatory Guide DG-1108 and decided to review the proposed final version of this Guide after resolution of public comments. The Committee has no objection to issuing DG-1108 for public comment.

Reference
Memorandum dated May 31, 2001, from Michael E. Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of Draft Regulatory Guide DG-1108, "Combining Modal Responses and Spatial Components in Seismic Response" -- Proposed Revision 2.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
K. Karwoski, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: RECOMMENDATION ON THE NEED TO REVISE 10 CFR PART 54, "REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR NUCLEAR POWER PLANTS"

Dear Chairman Meserve:

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," to resolve generic technical issues associated with license renewal. We also discussed this matter during our 483rd meeting on June 6-8, 2001. During our review, we had the benefit of the documents referenced.

Recommendation

10 CFR Part 54 is effective and efficient. It does not need to be revised at this time.

Discussion

In a Staff Requirements Memorandum (SRM) dated August 27, 1999, regarding SECY-99-148, "Credit for Existing Programs for License Renewal," the Commission asked the staff to prepare a detailed analysis and provide recommendations on whether it would be appropriate to resolve generic technical issues, including any credit for existing programs, by rulemaking. These recommendations were to be based on the accumulation of more data from license renewal applications of different designs and on experience gained from reviewing more applications.

Since the SRM was issued, the staff has reviewed license renewal applications for three pressurized water reactor plants and renewed their licenses. We have reviewed and commented on the Safety Evaluation Reports (SERs) associated with these applications. On the basis of our review, we believe that the license renewal process developed by the staff, with feedback from stakeholders, under the current rule is effective. This process is documented in a set of guidance documents: Generic Aging Lessons Learned (GALL) report, Standard Review Plan, and Regulatory Guide 1.188 that endorses NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule."

These guidance documents incorporate the resolution of technical issues, such as credit for existing programs, thus making the license renewal process understandable and predictable. Future updates of the guidance documents will provide the means for incorporating the resolution of remaining outstanding technical issues without amending the rule. Although review of the first boiling water reactor application for Hatch, Units 1 and 2, has not been completed, resolution of the open items in the interim SER does not appear to require rulemaking.

License renewal applications and their reviews have become increasingly efficient with subsequent applications. We expect them to become even more efficient when licensees endorse the approaches suggested by the now-approved guidance documents. Avoiding rulemaking at this time will further stabilize the existing process and facilitate the submittal and review of future applications.

Sincerely,

George E. Apostolakis
Chairman

<u>References</u>

1. Memorandum dated August 27, 1999, from Annette L. Vietti-Cook, Secretary, to William D. Travers, Subject: SECY-99-148 - Credit for Existing Programs for License Renewal.
2. Letter dated June 4, 2001, from Douglas J. Walters, Nuclear Energy Institute, to Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, Subject: License Renewal Rulemaking.
3. Letter dated June 26, 2001, from David Lochbaum, Union of Concerned Scientists, to Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, Subject: License Renewal Rulemaking.
4. Letter dated April 13, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Final License Renewal Guidance Documents.
5. Letter dated November 15, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: License Renewal Guidance Documents.
6. U. S. Nuclear Regulatory Commission, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated March 1, 2001.
7. U. S. Nuclear Regulatory Commission, NUREG-1801, Vols. 1 and 2, "Generic Aging Lessons Learned (GALL) Report," dated March 1, 2001.
8. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," March 2001.
9. Nuclear Energy Institute, NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," March 2001.
10. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report With Open Items Related to the License Renewal of Edwin I. Hatch, Units 1 and 2," February 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT NUREG-1742, "PERSPECTIVES GAINED FROM THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) PROGRAM"

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we reviewed the draft NUREG-1742, Vols. 1 and 2. During this review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced. Our Subcommittee on Reliability and Probabilistic Risk Assessment discussed this matter on June 22, 2001.

We agree with the staff that the IPEEE Program has been generally successful in meeting the intent of Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and has had a positive impact on safety.

Most IPEEEs are not based on probabilistic risk assessments (PRAs) and cannot be used in risk-informed regulatory decisionmaking. Most licensees chose to employ screening analyses and, consequently, the major external-event contributors to risk cannot be identified. Even with this limitation, the IPEEEs have confirmed the findings of earlier PRAs that earthquake- and fire-initiated accident sequences are important contributors to risk. This demonstrates the need for external-event PRAs to implement risk-informed regulatory decisionmaking.

The IPEEE results demonstrate the importance of human performance. Station blackout with failure to align and initiate the steam-driven auxiliary feedwater pump is one of the dominant earthquake-induced sequences for pressurized water reactors. For boiling water reactors, the dominant sequences following an earthquake include the failure of manual actions to recover power. Several manual recovery actions appear in risk-significant fire-initiated sequences, such as starting diesel generators and opening motor-operated valves when fire-induced cable damage occurs.

The report states correctly that there is no strong technical basis for the human action probabilities that are used in the IPEEEs. There is a clear need to improve the methodology used in the evaluation of the probability of unsatisfactory human performance. This should be identified as an important methodological issue for fire- and earthquake-initiated sequences.

We will comment on the unresolved safety issues and the generic safety issues that were addressed by the IPEEE Program after the staff responds to public comments on the report.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1742, Vols. 1 and 2, "Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program," Draft Report for Public Comment, April 2001.
2. U.S. Nuclear Regulatory Commission, NRC Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated June 28, 1991.
3. U. S. Nuclear Regulatory Commission, NRC Generic Letter No. 88-20, Supplement 5, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated September 8, 1995.
4. ACRS Report dated June 6, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Potential Use of IPE/IPEEE Results to Compare the Risk of the Current Population of Plants With the Safety Goals.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 23, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY REQUESTS FOR EXEMPTION TO EXCLUDE CERTAIN COMPONENTS FROM THE SCOPE OF SPECIAL TREATMENT REQUIREMENTS REQUIRED BY REGULATIONS (OPTION 2)

Dear Chairman Meserve:

During the 482nd, 483rd, 484th meetings of the Advisory Committee on Reactor Safeguards, May 10-11, June 6-8, and July 11-13, 2001, respectively, the Committee met with representatives of the NRC staff, the South Texas Project Nuclear Operating Company (STPNOC), and the Nuclear Energy Institute (NEI) to discuss STPNOC's requests for exemptions from certain Nuclear Regulatory Commission (NRC) regulations, including the adoption of a risk informed approach to the categorization and treatment of structures, systems, and components (SSCs). This matter was also discussed during a February 21, 2001, meeting of the Plant Operations and the Reliability and Probabilistic Risk Assessment Subcommittees. We also had the benefit of the documents referenced.

Recommendation

We concur with the staff that the STPNOC exemption requests should be granted as recommended in the preliminary safety evaluation report dated June 5, 2001.

Discussion

NRC regulations contain special treatment requirements that impose controls to ensure the quality and reliability of SSCs that are safety-related, important to safety, or otherwise within the scope of the regulations. These special treatment requirements include quality assurance (QA), environmental and seismic qualification, inspection and testing, and performance monitoring.

STPNOC has requested exemption from regulatory requirements for some SSCs. STPNOC has categorized SSCs based on risk rather than using the regulatory definition of basic components as found in 10 CFR 21.3.

The staff has determined that some requests for exemption should be granted and some should be denied, as follows:

Exemptions to be Granted

- 10 CFR 21.3 - Definition of Basic Component
- 10 CFR 50.34(b)(10) and (11) - Related to 10 CFR Part 100, Appendix A
- 10 CFR 50.49(b) - Scope of Electrical Equipment Important to Safety
- 10 CFR 50.55a(f) - ASME Inservice Testing
- 10 CFR 50.55a(g) - ASME Repair/Replacement & Inspection
- 10 CFR 50.55a(h) - IEEE 279 Quality & Qualification Requirements
- 10 CFR 50.59 - Changes, Tests, & Experiments
- 10 CFR 50.65(b) - Scope of Maintenance Rule
- 10 CFR Part 50, Appendix B - Quality Assurance Criteria
- 10 CFR Part 50, Appendix J - Type C Containment Leak Testing
- 10 CFR Part 100, Appendix A, VI, (a)(1) & (2) - SSE and OBE Design

Exemptions to be Denied

Licensee's Proposed Approach Meets Regulations:

- GDC 1 - Quality Standards and Records
- GDC 2 - Protection Against Natural Phenomena
- GDC 4 - Environmental and Dynamic Effects
- GDC 18 - Inspect/Test Electrical Power Systems

Update to QA Program Required to Reflect Changes to Commitments:

- 10 CFR 50.34(b)(6)(ii) - Appendix B Information Included in FSAR
- 10 CFR 50.54(a)(3) - Changes to QA Program

STPNOC categorized all SSCs in 29 safety systems into four risk categories according to their risk ranking. This ranking was based on a categorization process that used Probabilistic Risk Assessment (PRA) measures and the deliberations of an expert panel. The four risk categories were defined as:

- RISC-1 - Safety-Related and Risk-Significant
- RISC-2 - Non-Safety-Related and Risk-Significant
- RISC-3 - Safety-Related and not Risk-Significant
- RISC-4 - Non-Safety-Related and not Risk-Significant

No changes in regulatory treatment are proposed for SSCs that fall in categories RISC-1 and RISC-4.

STPNOC has committed to upgrade the 372 SSCs in the RISC-2 category to safety-related to the extent possible. This upgrade will result in an improvement in the safety posture of the facility.

The category that is of immediate interest is RISC-3, which contains the SSCs that are categorized as safety-related under the current system, but are not considered to be risk-significant using the STPNOC methodology. The question is: What treatment should be applied to these components?

STPNOC proposed that SSCs in RISC-3 be treated in accordance with "commercial practice" rather than the special treatment requirements currently applicable to safety-related components. Additional requirements were added to the Final Safety Analysis Report (FSAR) to help ensure the functionality of these components.

STPNOC has developed a methodology that uses PRA importance measures as inputs to the expert panel's structured decision-making process for categorizing SSCs.

STPNOC has developed a state-of-the-art PRA in which the licensee, the regulators, and the public can have confidence. The staff engaged an independent contractor to perform a review of the STPNOC PRA and their report indicates that the STPNOC PRA is of good quality.

Regulatory Guide 1.174 requires that the changes in CDF and LERF be small. There are no models that assess the impact of special treatment requirements on SSC failure probabilities. STPNOC performed a comparative risk analysis by increasing by a factor of 10 the failure rate of the RISC-3 components to be exempted from special treatment. STPNOC justified this choice of increased failure rate through an analysis of component failure data for safety-and non-safety-related components using data from the Nuclear Plant Reliability Data System (NPRDS), the Equipment Performance Information Exchange (EPIX), and the Maintenance Rule and Reliability Information (MRRI) database. The data showed that failure rates of components with commercial treatment were equivalent to failure rates of components with special treatment. The licensee argued that the choice of a factor of 10 was bounding although the failure data only apply to failures under normal operating conditions. Subsequently, the licensee compared the change in CDF and LERF to the original values and concluded the changes were acceptably small. STPNOC has also assessed the potential effect on the probability of late containment failure. It was also acceptably small.

Only about 6 percent of the SSCs currently classified as safety-related (or about 2,400 of the 44,000 SSCs in safety systems of the two units) are categorized based on PRA importance measures. The remaining 94 percent were analyzed and categorized by the expert panel. The deterministic method used by STPNOC's expert panel assured that the necessary instrumentation and controls upon which the operator may rely in emergency and severe accident conditions were categorized as risk-significant.

We have found the application by STPNOC to be adequate. Because plant-specific considerations are so important, the STPNOC application may not be an adequate template for similar applications by other licensees.

Mr. Stephen Rosen did not participate in the Committee's deliberations regarding this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated July 3, 2001, from M. A. McBurnett, STP Nuclear Operating Company, to NRC, Subject: Revised Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations.
2. Letter dated June 5, 2001, from S. A. Richards, NRC, to W. T. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2 - Factual Errors or Omissions in Preliminary Safety Evaluation on Exemptions Requested from Special Treatment Requirements.
3. Letters dated January 15, 18, and 23, March 19, May 8, and 21, 2001, January 29 and August 31, 2000, and October 14 and 22, 1999, from J. J. Sheppard, STP Nuclear Operating Company, to NRC Document Control Desk, Subject: Revised Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations.
4. Letter dated November 15, 2000, from J. Zwolinski, NRR, to W. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2 - Draft Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100.
5. Letter dated July 19, 2000, from J. A. Zwolinski, NRC, to W. T. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2 - Draft Review Guidelines on Risk-Informed Exemptions from Special Treatment Requirements.
6. Safety-Related Versus Non-Safety-Related Equipment Failure Frequency Data Analysis for Nuclear Power Plants in the United States, Final Report, South Texas Project Electric Generating Station, STP Nuclear Operating Company, April 6, 2000.
7. Letter dated January 18, 2000, from R. A. Gramm, NRC, to W. T. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2 - Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations, Request for Additional Information.
8. Letter dated July 13, 1999, from J. J. Sheppard, STP Nuclear Operating Company, to NRC Document Control Desk, Subject: Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 23, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: CIRCUMFERENTIAL CRACKING OF PWR VESSEL HEAD PENETRATIONS

Dear Chairman Meserve:

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we heard presentations by and held discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) Materials Reliability Program regarding industry and staff actions relative to cracking and leaking observed in pressurized water reactor (PWR) Alloy 600 reactor vessel head penetrations, including control rod drive mechanism (CRDM) nozzles. This matter was also discussed during a July 10, 2001, meeting of the Materials and Metallurgy and the Plant Operations Subcommittees. During our reviews, we had the benefit of the documents referenced.

Conclusions and Recommendations

1. The decision to issue a bulletin addressing the recent incidents of circumferential cracking of CRDM nozzles in U.S. PWRs is timely and appropriate.

2. The staff should urgently address technical issues associated with risk assessment, the effectiveness of inspection techniques, and the completeness of damage accumulation prediction.

Discussion

Cracks were recently detected during inspections of CRDM nozzles at Oconee Units 1, 2, and 3 and Arkansas Nuclear One (ANO) Unit 1. Preliminary risk assessment indicates that the issuance of a bulletin is appropriate to request operational information from the licensees as soon as possible.

The staff's in-depth analysis has raised a number of technical concerns. Although plans are in place to resolve them, the following concerns are of particular importance:

- Risk Assessment

  The risk assessment activities should be expanded to include rod ejection with coincident small-break loss of coolant accident and potential damage to adjacent control rods.

- Prioritization of Inspection Schedules

  Inspection schedule prioritization during the upcoming refueling outages will be based on an analysis of the susceptibility of cracking of CRDM nozzles in different plants. This approach relies on the assumption that susceptibility is determined by time of service and vessel head temperature. This has led to the grouping of each PWR into one of four "bins." The 14 reactors in the two highest susceptibility bins should receive highest priority in inspections of all CRDM nozzles in 2001. Although this approach is reasonable from a technical standpoint at present, its accuracy will become apparent as inspections proceed. It is prudent to consider potential modifications to this methodology including the following:

  (a) The cracking susceptibility will depend on other conjoint plant-specific factors that can affect cracking and that are not considered explicitly in the current susceptibility algorithm, which addresses only vessel head temperature and operating time. These further factors include residual stress, material composition, heat treatment, welding practices, and local chemical environment.

  (b) As more information on the cracking of CRDM nozzles accumulates from the upcoming U.S. inspections and from past observations overseas, the basis for a risk-informed methodology may be formulated.

  The staff should be prepared to modify any proposed inspection program and timing depending on the results of inspections of the first group of plants (i.e., Fall 2001).. These early inspection results may show that it is imperative to inspect the vessel heads of the remaining pressurized water reactors promptly. On the other hand, they may show that it is appropriate to delay the inspections of the remaining plants to allow improvements in diagnostic capabilities.

- Inspection Methods

  The current visual inspection process, which relies on detecting boron crystals at the top of the annulus, indicates the possible presence of circumferential cracks at the base of the annulus, but gives no information on the size and/or orientation of these cracks in the Alloy 600 material. In addition, the absence of visible boron crystals does not give complete assurance that a concentrated chemical environment at the annulus does not exist, resulting in the rapid growth of a circumferential crack. This concern could be addressed during the fall outage by a full volumetric inspection of all CRDM nozzles (i.e., including those with no boron crystals) at Oconee Units 1, 2, and 3, and ANO Unit 1. Volumetric inspections by a qualified process in such cases makes abundant sense. Assessment of the inspection methods used to detect and size cracks in CRDM nozzles and nozzle welds is necessary, especially for the circumferential cracks initiating at the base of the annulus between the CRDM nozzles and the pressure vessel head.

- Inspection Periodicity

  The inspection intervals once cracks are detected depend on knowledge of crack propagation rates as a function of the local material, environmental, and stress conditions. There are data for Alloy 600 cracking as a function of stress intensity and the temperature of the PWR primary coolant. Also, there are limited data relevant to the axial cracking in the Inconel 182 J-weld connecting the CRDM nozzle to the vessel head. The quality of these data is being evaluated by separate expert committees convened by industry and the staff. There is no similar data set relevant to the circumferential cracks that initiate in and adjacent to the J-weld and that present the greatest potential structural integrity concern. The reason for this lack of cracking data is that the local environment in the annulus between the pressure vessel and the CRDM nozzle is not known with sufficient certainty. This problem is also being addressed by the staff.

Consideration of the above issues in conjunction with the issuance of the bulletin should ensure that this matter is satisfactorily addressed for the short term. The Committee wishes to be updated once the licensee responses to the bulletin are evaluated.

A crucial issue confronted in the proposed bulletin is the urgency of inspections of vessel head penetrations, especially for plants thought to be less susceptible to CRDM stress corrosion cracking. Risk would be the metric best suited for determining the urgency. Unfortunately, neither the NRC's phenomenological capabilities, such as the ability to predict time-dependent stress corrosion cracking, nor the NRC's risk assessment capabilities are sufficiently developed at this time to provide defensible bases for decisions on the urgency of vessel head inspections. Sustained research to better the agency's integrated capabilities in probabilistic fracture mechanics and risk assessment will be needed to assist NRC in confronting future issues of reactor coolant system degradation.

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

Sincerely,

George E. Apostolakis
Chairman

References:
1. Letter dated June 29, 2001, from A. Marion, Nuclear Energy Institute, to Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: Response to June 22, 2001, letter from Dr. Brian Sheron (NRC) to Mr. Alex Marion (NEI) transmitting NRC staff questions on EPRI Interim Report TP-1001491, Part 2 (Proprietary).
2. U. S. Nuclear Regulatory Commission Proposed Bulletin 2001-XX, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated June 25, 2001.
3. Memorandum dated June 21, 2001, from C. E. Carpenter, Office of Nuclear Reactor Regulation, NRC, to W. Bateman, Office of Nuclear Reactor Regulation, NRC, Subject: Summary of June 7, 2001, Meeting with the EPRI Materials Reliability Program on Generic Activities Related to CRDM Cracking.
4. U. S. Nuclear Regulatory Commission Information Notice 2001-05: "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," dated April 30, 2001.
5. Electric Power Research Institute, TP-1001491, Part 2, "PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44)," Interim Report, May 2001.
6. Letter dated April 17, 2001, from Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, to Alex Marion, Nuclear Energy Institute; Subject: Issues to be Addressed in a Generic Justification for Continued Operation of PWRs.
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.
8. Letter dated June 1, 2001, from Alex Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, regarding NRC's Assessment of Topical Report MRP-44 - Summary of NRC/NEI Telecon of May 30, 2001.
9. Letter dated May 18, 2001, from Alexander Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: PWR Reactor Pressure Vessel Head Penetrations, dated, May 18, 2001.
10. Letter dated December 11, 1998, from David Modeen, Nuclear Energy Institute, Subject: Responses to NRC Requests for Additional Information on Generic Letter 97-01.
11. U. S. Nuclear Regulatory Commission Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, dated April 1, 1997.
12. Letter dated November 19, 1993, from William Russell, Office of Nuclear Reactor Regulation, NRC, to W. Rasin, Nuclear Utility Management and Resources Council (now NEI), transmitting the Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking.
13. Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, March 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 24, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SECY-01-0100, "POLICY ISSUES RELATED TO SAFEGUARDS, INSURANCE, AND EMERGENCY PREPAREDNESS REGULATIONS AT DECOMMISSIONING NUCLEAR POWER PLANTS STORING FUEL IN SPENT FUEL POOLS"

Dear Chairman Meserve:

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we met with representatives of the NRC staff and the Nuclear Energy Institute to review SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools." During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Safety Goals are appropriate for application to decommissioning plants with stored spent fuel.

2. Over the next year, the staff should prepare a strategy for developing and utilizing a safeguards probabilistic risk assessment (PRA) that includes the frequency of various levels of threats for both decommissioning and operating plants.

3. The requirement for emergency preparedness (EP) as a defense-in-depth measure should be maintained until there is a better technical basis for estimating the amount of warning time available.

4. The staff should initiate a study to determine the conditional probability of a zirconium fire as a function of decay time.

DISCUSSION

Policy Issues

In SECY-01-0100, the staff identified five policy issues along with a number of options for addressing these issues, and recommended a preferred option for each issue. Our views on these policy issues are as follows:

<u>Policy Issue #1: Should the Safety Goals for operating nuclear power plants be applied to decommissioning plants?</u>

The staff recommends that the Safety Goals for operating nuclear power plants be applied to decommissioning plants while spent fuel is being stored in the spent fuel pool (SFP). The rationale for this recommendation is that "spent fuel accidents can have public health and safety consequences similar to a core damage accident with a large offsite release."

We agree with this position and offer the following additional rationale: The Safety Goals can be thought of in terms of a risk/benefit concept. They represent acceptable risk to the public given the benefit of nuclear power generation. Since it is impossible to have the benefit of nuclear power without having decommissioning and stored spent fuel (DSF), the risk associated with DSF is part of the overall risk associated with nuclear power. Thus, the acceptable risk (i.e., the Safety Goals) should also be the same as for operating plants.

<u>Policy Issue #2: Should the Commission develop an approach using probabilistic risk assessment for quantifying the likelihood of sabotage that would permit greater risk-informed regulatory decision making in the area of safeguards?</u>

<u>Policy Issue #3: How should the Commission define the safeguards protection goal to be applied to SFPs at decommissioning plants?</u>

Our views on policy issues #2 and #3 are provided below.

The staff recommends that the new regulatory requirements for safeguards at decommis-sioning plants be based on deterministic and performance criteria (Option 3). The recom-mended safeguard goal consists of a design criterion of protecting against radiological sabotage by the design basis threat and a performance standard of limiting radiation dose to an individual to 5 rem at 100 meters. While the staff is not recommending the development of a risk-informed approach for quantifying the likelihood of sabotage, it pledges to continue to look for opportunities to increase the use of PRA technology in the safeguards area.

Given the current state of the art with respect to a safeguards PRA, we agree with the staff's recommended Option 3. It is not premature for the Office of Nuclear Regulatory Research to have a program to develop a safeguards PRA (Option 2). Such a PRA will need to include the frequencies of various levels of threats, the conditional probability that a specified threat will lead to spent fuel uncovery with various resulting geometries, and the associated consequences. We recognize that these needs are significantly different and more challenging than for a standard PRA, particularly with respect to the initiating-event frequencies. Nevertheless, we believe there would be substantial benefit in developing a safeguards PRA for both decommissioning and operating plants. For spent fuel pools, a particular need is to determine the conditional probability of a zirconium fire given various threats. Over the next year, the staff should prepare a strategy for developing and utilizing a safeguards PRA that includes the frequency of various levels of threats for both decommissioning and operating plants.

Policy Issue #4: What level of insurance is appropriate for licensees of decommissioning plants given the low likelihood of a large onsite and offsite radiological release from a zirconium fire accident involving the spent fuel stored in the SFP?

The staff maintains that the risk associated with SFP accidents will be kept low. The staff believes that the contribution from sabotage can be kept low by protecting the spent fuel against the design basis threat. Therefore, the staff recommends that insurance requirements be substantially reduced shortly after a reactor permanently shuts down and enters into decommissioning.

We agree that the level of required insurance should be commensurate with the level of risk. We accept the staff's assessment that the level of initial reduction of the required insurance is appropriate. If the intent, however, is to further incrementally reduce the level of insurance, the large uncertainties associated with the threats from sabotage will have to be addressed. Once again, this calls for a safeguards PRA that includes the conditional probability of a zirconium fire as a function of decay time.

Policy Issue #5: What level of offsite emergency preparedness is appropriate for decommissioning plants given the low likelihood of a radiological release large enough to exceed protective action guides offsite?

The staff recommends that offsite EP be reduced incrementally and eventually eliminated. The staff's rationale is based on a conclusion of NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," that the risk of a zirconium fire at a decommissioning plant is well below the Commission's Quantitative Health Objectives. The staff also concludes that, a few months after shutdown, the contribution of offsite EP to reducing overall risk is small for the accident sequences analyzed and that the risk change from relaxing EP is within the guidelines of Regulatory Guide 1.174.

The staff may be right in this assessment, but there is considerable uncertainty associated with the risk contribution due to sabotage. We have previously argued that defense-in-depth should be applied in areas where the risks and the associated uncertainties could be large. We think these attributes well describe the risk associated with decommissioning plants and this appears to be a place to maintain an appropriate level of EP as defense-in-depth. If a defensible technical basis can be developed for concluding that the probability of a zirconium fire is acceptably low after some reasonable decay time, this would be a sufficient basis for relaxing the EP requirements. For example, the staff states that, given a warning time on the order of 10 hours, ad hoc emergency response measures would be effective. We agree, but believe that a more definitive technical assessment is needed to determine the warning time that would be available as a function of decay time.

Existing Exemptions

Several plants undergoing decommissioning were granted exemptions to EP and insurance requirements, in part because of the belief that a zirconium fire was not possible given the decay time that had elapsed for these plants. One of the conclusions of NUREG-1738, however, is that a zirconium fire cannot be dismissed even many years after shutdown. Nevertheless, the staff judges that previously granted exemptions for EP and insurance at currently decommissioning plants do not present an undue risk to the public health and safety, given the long time periods available to support implementation of protective or mitigative measures on an ad hoc basis for SFP accidents.

We agree that there is not an immediate undue risk to the health and safety of the public based on the long times associated with SFP accidents and the presumed low likelihood of sabotage events.

Staff And Industry Differences

Representatives of the industry disagree with various findings in NUREG-1738. They differ with the report's conclusions that a zirconium fire could not be dismissed even after many years of decay time, that the ruthenium release would be substantial, and the probability that cask drop events could lead to rapid draining of the spent fuel pool. We believe these differences could be resolved, although they do not impact the decisions made on the options discussed here.

The industry representatives recommend that NUREG-1738 be subjected to a formal peer review. We would prefer a peer review on our recommended study related to the probability of a zirconium fire.

Sincerely,

George E. Apostolakis
Chairman

References:

1. SECY-01-0100, dated June 4, 2001, memorandum from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Policy Issues Related to Safeguards, Insurance, And Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools.
2. U.S. Nuclear Regulatory Commission, NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued January 2001.
3. Report dated April 13, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.
4. Report dated November 8, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.
5. Letter dated January 18, 2001, from William D. Travers, Executive Director for Operations, NRC, to George E. Apostolakis, Chairman, ACRS, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.

6. Letter dated April 26, 2001, from Ralph E. Beedle, Nuclear Energy Institute, to Samuel J. Collins, Office of Nuclear Reactor Regulation, NRC, Subject: NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants."
7. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
8. J. N. Sorensen, G. E. Apostolakis, T. S. Kress, and D. A. Powers, "On The Role of Defense In Depth in Risk-Informed Regulation," presented at PSA '99, Washington, D.C, August 22-25, 1999.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 25, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: FEASIBILITY STUDY ON RISK-INFORMING THE TECHNICAL REQUIREMENTS OF 10 CFR 50.46 FOR EMERGENCY CORE COOLING SYSTEMS

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we met with representatives of the NRC staff and the industry to discuss the status of staff and industry initiatives to risk inform the technical requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." Our Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment discussed this matter with representatives of the NRC staff, the Nuclear Energy Institute, the Westinghouse Owners Group, and the Boiling Water Reactor Owners Group on July 9, 2001. We also had the benefit of the documents referenced.

Recommendations

1. We recommend that the Commission approve the staff's request to proceed with rulemaking to modify the existing 10 CFR 50.46 to replace the prescriptive emergency core cooling system (ECCS) acceptance criteria with a performance-based requirement and to modify the 10 CFR Part 50, Appendix K evaluation model.

2. We recommend that the Commission approve the staff's request to proceed with the development of a voluntary risk-informed alternative to 10 CFR 50.46, Appendix K, and General Design Criterion (GDC) 35 of 10 CFR Part 50, Appendix A.

3. The staff should continue to develop the technical bases and requirements for redefining the large-break loss-of-coolant accident (LBLOCA).

Discussion

The ECCS requirements codified in 10 CFR 50.46, Appendix K, and GDC 35 are intended to ensure that plants can safely cope with a LBLOCA. The ECCS has been designed to accommodate pipe breaks up to and including a double-ended guillotine break of the largest pipe in the reactor coolant system. GDC 35 requires that the ECCS be capable of providing sufficient core cooling for a full spectrum of postulated LOCAs using either offsite power or onsite power. To comply with this requirement, ECCS evaluations generally assume that pipe breaks are coincident with a loss of offsite power (LOOP). In addition, the system must have sufficient diversity and redundancy to accomplish its safety function assuming a single failure.

Because LBLOCAs are rare, the current requirements for ECCS performance may have a detrimental effect on safety. These requirements focus attention and resources on events that are extremely unlikely to happen rather than on events which can have a larger contribution to risk. For example, the postulated occurrence of a LOOP coincident with a LBLOCA leads to requirements for rapid emergency diesel generator (EDG) start times and load sequencing. Such requirements could reduce the reliability of the EDGs and diminish the capability of the system to deal with the more likely small and medium break LOCAs.

The industry has proposed a revision of 10 CFR 50.46 that is based on a redefinition of the LBLOCA. Instead of dealing with a full spectrum of break sizes up to and including the double-ended guillotine break of the largest pipe in the reactor coolant system, the industry proposes to define a new maximum LBLOCA size based on leak-before-break (LBB) methodology and probabilistic assessments of the frequency and consequences of the new LBLOCA size.

The staff has accepted LBB methodology for the analysis of dynamic effects of pipe failure for pipe sizes down to 8-inches in some cases. The NRC pioneered the application of probabilistic fracture mechanics to piping through the development of the PRAISE code. The staff argues, however, that the prediction of leak rates for all sizes of cracks in all locations in piping systems is technically much more demanding than predicting whether a detectable leak will occur before failure. The staff also argues that a more rigorous assessment of uncertainties is needed to justify the redefinition of the LBLOCA for ECCS requirements. Thus, the staff believes this is a longer-term activity that will require a substantial technical effort.

We agree that the effort to define a new LBLOCA size requires an extension of current LBB and probabilistic fracture mechanics methodology. We believe that it is technically feasible, but the justification of the new LBLOCA size will become increasingly difficult as the proposed maximum break size is decreased. The industry has stated that it is willing to invest substantial resources to accomplish this objective. The staff should continue to develop the technical basis and requirements for the redefinition of LBLOCA.

In its Feasibility Study, the staff has investigated a number of options for revising 10 CFR 50.46 that it believes can be implemented on a shorter time scale and will provide safety benefits and some reduction in unnecessary conservatism and associated regulatory burden.

One of these options would make changes in the Appendix K evaluation model and would replace the current prescriptive ECCS acceptance criteria with a performance-based requirement. This would permit licensees to use cladding materials other than zircaloy or ZIRLO without having to seek an exemption. The current criteria, such as the 2200°F peak clad temperature and 17% oxidation limit, would be relegated to a regulatory guide as acceptance criteria for zircaloy and ZIRLO. We support the proposed development of the new performance-based acceptance requirement.

Possible changes in the evaluation models suggested in the staff Feasibility Study include replacing the current 1971 American Nuclear Society (ANS) decay heat curve with the 1994 ANS standard, replacing the current decay heat multiplier of 1.2 with an uncertainty estimate, and replacing the Baker-Just oxidation model with the Cathcart-Pawel oxidation model for heat generation. The intent of these changes is to use improved technical understanding to remove excessive conservatism from Appendix K models.

We are generally supportive of this effort, but note that in dealing with a mix of models in which some elements are conservative and some elements are nonconservative, removing "excessive" conservatism without a real understanding of the uncertainties in the overall model can lead to unsatisfactory results. For example, although the Cathcart-Pawel model gives a more accurate description of the oxidation behavior of unirradiated zircaloy tubing in laboratory studies, the more conservative Baker-Just model was deliberately chosen in an attempt to ensure that the effects of variables such as irradiation and behavior such as spalling of the oxide film that were not explicitly included in the models would not lead to nonconservative results. In addition, although the staff is developing performance-based acceptance criteria to permit use of other cladding materials, both the Baker-Just and Cathcart-Pawel models build "zircaloy behavior" into the evaluation model. The staff should consider a performance-based requirement for a heat generation model that includes the effects of cladding oxidation, irradiation, and the potential for cladding spallation rather than a prescriptive requirement. Acceptable heat generation models for different cladding materials could then be discussed in a regulatory guide. If implementation of the Appendix K option proves to be more challenging than anticipated, then the staff should proceed with a rulemaking that includes only the update of the decay heat curve to the 1994 ANS standard.

The second shorter-term option recommended by the staff is a voluntary risk-informed alternative to 10 CFR 50.46 that would replace the current requirements intended to ensure ECCS reliability (i.e., the coincident LOOP and the single-failure criterion) with more risk-informed approaches that reflect the lower frequencies of LBLOCAs. Licensees could choose either generic deterministic reliability requirements developed by the NRC (e.g, a requirement that a coincident LOOP be postulated only for smaller, more frequent LOCAs) or show that they can meet an acceptable threshold value for the core damage frequency (CDF) and large, early release frequency (LERF) associated with the LOCA initiators with appropriate consideration of uncertainties. ECCS reliability evaluations could reflect plant-specific features and operational data. Frequencies of LOCAs with different break sizes could be determined using the analysis provided in NUREG/CR-5750, updated to reflect more recent operating experience. Alternatively, probabilistic fracture mechanics together with a review of service history data could be used, but the technical work to support this would be similar in magnitude to that required to define the new LBLOCA size. We believe the approach outlined by the staff in this option would provide a much more realistic and risk-informed approach for ECCS requirements. The staff should proceed with the technical work and the rulemaking for this option.

We look forward to reviewing the technical work and regulatory guidance needed to support these rulemaking efforts as they evolve.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Draft memorandum received June 3, 2001, from William D. Travers, Executive Director for Operations, to The Commissioners, Subject: Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria), and attached Feasibility Study report.
2. Memorandum dated January 19, 2001, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-00-0198 - Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes 10 CFR 50.44 (Combustible Gas Control).
3. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
4. Letter dated February 8, 2001, from Anthony R. Pietrangelo, Nuclear Energy Institute, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Preliminary Industry Response to NRC Questions on Redefinition of Large-Break Loss-of-Coolant Accident.
5. Letter dated October 17, 2000, from Robert H. Bryan, Westinghouse Owners Group to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: "WOG Large Break Loss of Coolant Accident (LBLOCA) Redefinition Discussion of Benefits."
6. Letter dated January 8, 2001, from Adrian Heymer, Nuclear Energy Institute, to Mary T. Drouin, Office of Nuclear Regulatory Research, Subject: "Draft Large Break LOCA Redefinition Program, Project Summary."
7. Letter dated January 19, 2000, from Joe F. Colvin, Nuclear Energy Institute, to Richard A. Meserve, Chairman, NRC, Subject: SECY-99-264, Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
8. American Nuclear Society, ANSI/ANS-5.1-1994, American National Standard for Removing Decay Heat Power in Light Water Reactors, dated August 23, 1994.
9. U.S. Nuclear Regulatory Commission, NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, February 1999.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 25, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

Sam Duraiswamy /For

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDE (DG)-1077, "GUIDELINES FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED EQUIPMENT IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS"

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, the Committee considered the draft Regulatory Guide DG-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety in Nuclear Power Plants," and decided to review it after reconciliation of public comments. The Committee has no objection to issuing the draft Regulatory Guide for public comment.

Reference
Memorandum dated June 8, 2001, from Sher Bahadur, Office of Nuclear Regulatory Research, to John Larkins, Executive Director, ACRS, Subject: Draft I&C Qualification Regulatory Guide Package DG-1077, Version 1.7

cc: A. Vietti-Cook SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
M. Mayfield, RES
S. Arndt, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 25, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Sam Duraiswamy /For
John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: WITHDRAWAL OF REGULATORY GUIDE 1.120, "FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS"

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, the Committee decided that it has no objection to the staff's proposal to withdraw Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants."

Reference:
E-Mail dated July 11, 2001, to Sher Bahadur, ACRS, from John Hannon, NRR, Subject: Request for a "No Objection Letter for the Withdrawal of Regulatory Guide 1.120."

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
S. Collins, NRR
E. Connell, NRR
J. Hannon, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 13, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED FINAL REVISION TO REGULATORY GUIDE 1.78, "EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE"

Dear Dr. Travers:

During the 485th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 2001, we reviewed the proposed final revision to Regulatory Guide 1.78, "Evaluating The Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Revision 1 to Regulatory Guide 1.78 is to ensure habitability of a reactor control room in the event of an accident off-site or onsite involving the atmospheric dispersal of toxic chemicals. This revision incorporates and withdraws Regulatory Guide 1.95 concerning accidental chloride releases, since many regulatory positions in these two guides are the same or similar. The staff has formulated this revision to make it less prescriptive and more performance oriented. The revision also allows licensees to use the results of quantitative risk analyses in their evaluations.

Plants vary widely in their vulnerabilities to accidents involving atmospheric releases of toxic chemicals that might be drawn into reactor control rooms. The revised Regulatory Guide 1.78 provides conservative screening criteria well founded on recent standards for toxic chemical concentrations that are considered "immediately dangerous to life and health." The screening criteria also recognize site-specific weather conditions and control room leakage.

In the revised Regulatory Guide, the staff offers three options to further examine the threats of toxic chemical releases on control room habitability. Licensees may:

- Use arguments based on quantitative assessments of risk.
- Adopt the performance criteria defined in the revised Regulatory Guide.
- Use prescriptive analyses and measures accepted by the staff in the past.

The treatment of performance-based approaches in the revised Regulatory Guide is of particular interest. The staff has defined criteria based on toxic chemical concentrations in the control room and permits licensees to use technically justifiable means to show that they meet these criteria for control room habitability. The licensees are asked to address both maximum concentration accidents (short-term instantaneous releases) and maximum concentration-duration accidents (long-term, low-leakage-rate releases). In addition the licensees are to evaluate atmospheric dispersion, control room air flow, detection systems, control room isolation systems, personnel protection systems, and emergency planning. The revised Regulatory Guide suggests possible methods of analysis acceptable to the staff. Alternatively, licensees may use quantitative risk arguments, as described in Regulatory Guide 1.174.

The revised Regulatory Guide 1.78 should be issued for use by licensees. It should improve safety as well as reduce burdens on both licensees and the staff. Furthermore, this revised Guide provides a good example of how regulatory guides may be made more performance oriented.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated August 9, 2001, from Thomas L. King, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Prepublication Copy of the Regulatory Guide 1.78, Revision 1 (Previously Issued as DG-1087 for public comment).
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chloride Release," issued January 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 13, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDES CONCERNING CONTROL ROOM HABITABILITY, DOSE ASSESSMENT, METEOROLOGICAL ASSESSMENT, AND TESTING

During the 485th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 2001, the Committee considered the four referenced draft regulatory guides related to main control room habitability. The Committee plans to review the proposed final version of these regulatory guides following the reconciliation of public comments. The Committee has no objection to issuing these guides for public comment.

<u>References</u>:

1. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 1.control room habitability, "Control Room Habitability at Nuclear Power Reactors," received September 6, 2001.
2. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 1.test, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," received September 6, 2001.
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," received September 6, 2001.
4. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Nuclear Power Reactors," received September 5, 2001.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
G. Holahan, NRR
J. Hayes, NRR
M. Hart, NRR
S. La Vie, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 13, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED FINAL REVISIONS TO REGULATORY GUIDES 1.142, "SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS," AND 1.143, "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS"

During the 485th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 2001, the Committee considered the proposed final revisions to the subject regulatory guides and decided not to review them. The Committee has no objection to the issuance of these revised regulatory guides.

Reference:

1. Memorandum dated August 3, 2001, from Michael E. Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Request for Review and Concurrence to Issue Proposed Revision 2 to Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants," (Formerly DG-1098) and Proposed Revision 2 to Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Formerly DG-1100).

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
M. Mayfield, RES
H. Graves, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 14, 2001

Dr. William D. Travers
Executive Director for Operations
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: GENERIC SAFETY ISSUE-191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PUMP PERFORMANCE"

During the 484th and 485th meetings of the Advisory Committee on Reactor Safeguards, July 11-13 and September 5-7, 2001, we heard presentations by and held discussions with representatives of the NRC staff regarding the Office of Nuclear Regulatory Research recommendation for resolving Generic Safety Issue (GSI)- 191, "Assessment of Debris Accumulation on PWR Sump Pump Performance." During this review, we had the benefit of the documents referenced.

We agree with the staff that potential issues associated with the performance of pressurized water reactor containment sumps have been identified. The NRC staff should expeditiously resolve GSI-191. If plant-specific analyses are required as part of the resolution, guidance for performing these analyses should be developed. We would like to review the proposed final disposition of this issue.

Sincerely,

George E. Apostolakis
Chairman

Reference:
Letter dated August 29, 2001, from Michael E. Mayfield, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: RES's Proposed Recommendation for Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," attaching:

(1) Rao, D., et al., "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," LA-UR-XXX, Los Alamos National Laboratory, Los Alamos, New Mexico, July 2001.

(2) Buslik, A., Risk Considerations and Benefits Associated with GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," U.S. NRC, August 8, 2001.

(3) Feld, S., Cost Analysis for GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," U.S. NRC, August 14, 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 17, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: APPLICATION OF GE NUCLEAR ENERGY TRACG CODE TO ANTICIPATED OPERATIONAL OCCURRENCES

During our 485th meeting on September 5-7, 2001, the Advisory Committee on Reactor Safeguards met with representatives of the NRC staff and GE Nuclear Energy (GE) to review the application of the GE TRACG realistic or "best-estimate" code to Anticipated Operational Occurrence (AOO) transient events. Our subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during meetings held on November 13-14, 2000, and August 22-23, 2001. During our review, we had the benefit of the documents referenced.

CONCLUSION

On the basis of our review, we support the staff's finding that Version 02A of the TRACG code is acceptable for application to AOO transients.

DISCUSSION

All thermal-hydraulic codes are approximations to reality; they are assembled with careful attention to the regulatory needs that they will be required to meet. This requires adequate modeling of the important physical phenomena that are usually identified by the Phenomena Identification and Ranking Table (PIRT) process. As such, these models must be reasonable and founded on sound technical principles and experimental evidence, but they are never perfect. Therefore, using such models to support regulatory decisions usually requires an assessment of the uncertainties in the predictions generated by the code.

A key condition for the acceptability of a realistic code is that there be adequate quantitative measures of this uncertainty in the predictions of regulatory parameters and success criteria used for probabilistic risk assessment. An acceptable method for assessing these uncertainties is described in NUREG/CR-5249, Quantifying Reactor Safety Margins -- Application of the

Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident (LOCA), dated December 1989.

GE has done an exemplary job of applying the CSAU methodology to Version 02A of the TRACG code, as it applies to AOO transients. These transients have potential importance for providing limits to power uprate and reducing requirements for emergency diesel generator start times, for example.

In the case of the TRACG code, GE assessed the code uncertainties on the basis of full-scale separate-effects tests, component performance data and full-scale plant data for boiling-water reactors (BWRs), as well as scaled integral tests. Confidence in the use of the code is also considerably enhanced by its ability to properly predict the AOO transients that have actually occurred in a number of plants of the various BWR types.

During the course of their reviews, our Thermal-Hydraulic Phenomena Subcommittee and the staff raised several questions about some models in the code. GE satisfactorily answered all of these questions. Even in cases where the Subcommittee found good reason to question some feature of a model, it was satisfied that use of the CSAU methodology enabled assessment of the significance of assumptions or approximations within the context of their use in evaluating AOO transients. While a better model might lead to less uncertainty, perhaps allowing a reduction in the level of conservatism implied in margins to be reduced, rational decisions can be made in the presence of uncertainty, as long as one knows the extent of that uncertainty. Thus, responding to the staff's comments on our letter of January 11, 2001, regarding the use of industry-developed thermal-hydraulic codes, we stated in our letter dated June 19, 2001, that the agency needs to decide how these quantitative measures of uncertainty can be used in a more formal way to support the rationale for regulatory decisions.

Our confidence in the acceptability of Version 02A of the TRACG code is supported by GE's professionalism, its willingness to respond openly to questions raised by the Committee and the staff, and its willingness to supply the source code to the staff.[1] In addition, the staff inspired confidence through its independent use of the code to assess its neutronic aspects. The staff's judgment was that the thermal-hydraulic features of the code had already been well assessed in previous reviews and did not need to be further evaluated by independent computer runs. Nonetheless, it will be advisable for the staff to perform its own computer runs to evaluate some thermal-hydraulic features of the code as part of future assessments of other transients, such as anticipated transients without scram and LOCA.

Another independent assessment that we consider to have significant positive influence on public confidence is the staff's assessment of the neutronic features of the code. In that assessment, the staff made comparisons with the predictions of the NRC's TRAC-B/NESTLE code. We look forward to similar comparisons with the thermal-hydraulic predictions of the TRAC-M code, which is nearing completion by the NRC's Office of Nuclear Regulatory Research, when the staff considers further applications of TRACG.

---

[1]We note that Section II.1.b. of Appendix K to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50) requires that "A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request."

ACRS members Mario Bonaca and F. Peter Ford did not participate in the Committee's review of this matter.

Sincerely,

George E. Apostolakis
Chairman

REFERENCES

1. U. S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, NEDE-32906P, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis," undated (Proprietary).
2. General Electric Nuclear Energy Licensing Topical Report, NEDE-32176P, "TRACG Model Description," Revision 2, December 1999 (Proprietary).
3. General Electric Nuclear Energy Licensing Topical Report, NEDE-32906P, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," January 2000 (Proprietary).
4. General Electric Nuclear Energy Licensing Topical Report, NEDE-32177P, "TRACG Qualification," Revision 2, January 2000 (Proprietary).
5. General Electric Nuclear Energy Licensing Topical Report, NEDE-32900P, "TRACG Licensing Application Framework for AOO Transient Analysis," June 1999 (Proprietary).
6. General Electric Nuclear Energy Licensing Topical Report, NEDC-32956P, "TRACG02A User's Manua,l," February 2000 (Proprietary).
7. Letter dated January 11, 2001, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Richard A. Meserve, Chairman, NRC, Subject: Issues Associated with Industry-Developed Thermal-Hydraulic Codes.
8. Letter dated June 19, 2001, from George E. Apostolakis, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Response to Your April 12, 2001 Letter on Issues Raised by ACRS Pertaining to Industry Use of Thermal-Hydraulic Codes.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 12, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Chairman Meserve:

During our 485th meeting on September 5-7, 2001, the Advisory Committee on Reactor Safeguards met with representatives of the NRC staff to discuss the revised Reactor Oversight Process (ROP). We continued our deliberations during our 486th meeting on October 4-6, 2001. This matter was also discussed during meetings of the ACRS Plant Operations Subcommittee on December 6, 2000, May 9, 2001, and July 9, 2001. In addition, the ACRS Subcommittees on Plant Operations and Fire Protection held meetings with licensees on June 13, 2000, and June 27, 2001, and held meetings with Regions III and IV on June 14, 2000, and June 28, 2001, respectively. During our review, we had the benefit of the documents referenced.

BACKGROUND

The ROP utilizes the results of performance indicators (PIs) and baseline inspection findings to determine the appropriate regulatory action to be taken in response to a licensee's performance. The escalation of the regulatory responses is specified in the action matrix, which the staff developed as part of the ROP. This ROP has been in effect for nearly all licensees for about one year. The staff has conducted an assessment of the state of the ROP and recognizes that it is still a process in development.

The ACRS has previously commented on various aspects of the ROP and provided recommendations to the staff regarding potential process improvements. We remain convinced that the ROP is more objective and understandable than the former oversight process and represents a significant improvement. This report discusses some specific questions that the Commission raised to the ACRS, and offers some additional thoughts on potential improvements in the ROP.

In the Staff Requirements Memorandum dated April 5, 2000, the Commission requested the ACRS to:

(1) Review the use of PIs in the ROP to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.

(2) Review the initial implementation of the significance determination processes (SDPs), and assess the technical adequacy of the SDP to contribute to the ROP.

The current PIs do provide meaningful insight into plant performance. However, there is a need to redefine the thresholds for some of the PIs to provide better input to the ROP. In particular, the numerical values for the white/yellow and yellow/red thresholds for the initiating event and mitigation system PIs are not useful and should be revised. The color bands for the PIs and SDPs associated with all the cornerstones have similar implications with respect to agency action and, therefore, the thresholds should be commensurate with their respective safety significance.

The most immediate and pressing need for the ROP is to improve the SDP tools. Some SDPs are incomplete and, in cases such as fire protection, overly subjective. The technical adequacy of the risk-based SDPs depends on the availability and quality of a relevant probabilistic risk assessment (PRA). Thus, the SDP for at-power situations provides meaningful risk information. For routine findings that are predominantly of very low, low, and moderate safety significance, the process is probably adequate. The threshold values for the risk-based SDPs are appropriate.

We continue to believe that a documented review of the SDP worksheets and SPAR models (as well as the underlying SAPHIRE computer code) is essential to public confidence in the ROP.

An SDP based on low-power and shutdown PRAs or other shutdown management tools is needed to characterize findings during these modes of operation. In addition, the fire protection SDP involves very qualitative inputs to a quantification process of uncertain pedigree. This SDP is probably useful for its intended purpose, however, it may be hard to defend and justify to the public. Even though this SDP calculates the change in core damage frequency (CDF), the SDP is really intended to provide an indication of the degradation of defense in depth for fire protection as defined in 10 CFR Part 50, Appendix R.

Presently, concurrent performance deficiencies are assessed collectively, as applicable, to determine the total change in CDF, but each performance deficiency is assigned a color individually. There may be instances in which conclusions could be altered if the results are considered collectively, and thus such collective results should be considered in the action matrix.

DISCUSSION

An important premise of the ROP is that there should be a graded regulatory response to inspection findings and PI results. Although a graded response to oversight findings is a desirable attribute, the inputs to the action matrix that implements this response must be produced in a way that justifies the resulting response. This is especially true for the right-hand columns of the matrix which could lead to severe regulatory responses.

The current ROP uses different technical bases to establish the thresholds for the PIs and inspection findings. In particular:

- On the basis of its review of recent operating history, the staff set the green/white thresholds for the PIs for initiating events and mitigating systems at the 95th percentile of peer performance for the given PI. By contrast, the staff based the white/yellow and yellow/red thresholds on an assessment of the value of a PI corresponding to increases in CDF of 10^{-5} and 10^{-4} per reactor year, respectively.

- The staff set the PI thresholds for barriers, emergency preparedness, occupational radiation safety, public radiation safety, and physical protection by considering technical specification limits, the number of noncompliances with regulatory requirements, and other absolute measures.

- The staff based the green/white, white/yellow, and yellow/red thresholds for SDP results on increases in CDF of 10^{-6}, 10^{-5}, and 10^{-4} per reactor year, respectively. This is true for the initiating event, mitigating system, and fire protection cornerstones. The other SDPs do not have a PRA basis and take a deterministic and defense-in-depth approach to establish thresholds for safety significant issues.

These different bases for defining the various thresholds raise questions regarding the kinds of information that the PIs and SDPs provide and the consistency of the meaning of the thresholds across the PIs and SDPs. These different thresholds are based on expert judgment that the degradation in performance associated with each color band is appropriately linked to a corresponding regulatory response[1].

It is from this viewpoint that we believe it is necessary to reconsider the definitions of the white/yellow and yellow/red thresholds for initiating events and mitigating systems, which as we noted above were based on an attempt to assess the value of a PI corresponding to increases in CDF.

We have noted previously that it is difficult to generically assess the risk impact of changes in a PI. The associated changes in risk tend to depend strongly on plant-specific features. This approach, however, has a deeper, more intractable flaw. Specifically, it focuses on the change in CDF that results from changes in a single, isolated parameter assuming that all other factors that can affect CDF remain constant. A realistic assessment of the change in CDF cannot be related to the change in a single PI. Thus, in some cases, the use of this approach to select white/yellow and yellow/red thresholds has led to values for these thresholds that, in our judgment and that of many of the staff and the industry, are too high to be meaningful. Regulatory attention would increase at much lower levels.

---

[1] The color bands for the ROP are called "constructed scales" in decision analysis. Ensuring the consistency of the bands of these scales is what decision analysts commonly call "performing sanity checks," and such checks are among the most important steps in a decisionmaking process. In our report on the NRC Safety Research Program (NUREG-1635, Vol. 4), we recommended that the staff initiate a program of research to investigate how best to use formal decisionmaking methods in regulatory decisions.

The white/yellow and yellow/red thresholds for the PIs for initiating events and mitigating systems should be set in terms of an expert judgment of what values should in fact trigger the regulatory response associated with the threshold. Although general considerations for the selection of thresholds for PIs and SDPs are discussed in SECY-99-007, the expert judgment process that the staff used to develop the initial values for the thresholds for the non risk-based PIs and SDPs and the corresponding equivalency of the combination of findings in the action matrix have not been well documented. The NRC has been a pioneer in the use of scrutable expert judgment processes, and it is unfortunate that the use of expert judgment in a process as central to the NRC's mission as the ROP lacks the traceability of other NRC uses of expert judgment. Formal decision analysis could be helpful in making the selection of thresholds and the action matrix more objective and scrutable.

In assessing the need to revise the current PIs and develop new PIs, we believe that the staff responsible for the ROP should consider the work being done in other parts of the agency. For example, the review of operating experience for the reactor core isolation cooling (RCIC) system for BWRs (NUREG/CR-5500, Vol. 7) shows that the dominant failure modes involve system failures while running and human failures to recover the system (i.e., failures that are not part of the unavailability calculations that the ROP requires). In analyzing the operating experience, the analysts distinguished between two contexts of RCIC system operation: (1) short-term missions (less than 15 minutes), in which the system must inject water into the reactor vessel following a scram with feedwater available and the main isolation valves open, and (2) long-term missions, in which the system must inject water into the reactor vessel following a scram with feedwater unavailable and/or the reactor vessel isolated. The average system unreliability in these two contexts differs by a factor of 2. The ROP green/white threshold for RCIC system unavailability is 0.04 and makes no distinction between the two contexts identified in the study driven by operating experience. Since unreliability is a metric that includes all potential failure modes, it should be included in the PIs.

We continue to believe that it is important that there be consistency in the definition of terms like "unavailability" which are used in the PIs. Inconsistencies in technical terms that the agency uses in several major activities make comparisons and communication, both internally and externally, difficult.

The ROP is an evolving process. The staff has done an excellent job establishing the basic framework in a relatively short period of time considering the scope of this project. We look forward to continued interactions with the staff on this very important matter.

Additional comments by ACRS Members George E. Apostolakis, Thomas S. Kress, and Steven L. Rosen are presented below.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Staff Requirements Memorandum dated April 5, 2000, from Annette L. Vietti-Cook, Secretary, NRC, to Dr. John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting on March 2, 2000, with ACRS on Risk Informing 10 CFR Part 50.
2. Letter dated March 15, 2000, from Dana A. Powers, ACRS Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Revised Reactor Oversight Process.
3. NRC Inspection Manual, Manual Chapter 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, February 5, 2001.
4. NUREG/CR-5500, Vol. 7, Reliability Study: Reactor Core Isolation Cooling System, 1987 - 1993, Idaho National Engineering and Environmental Laboratory, September 1999.
5. U. S. Nuclear Regulatory Commission, SECY-99-007, Recommendations for Reactor Oversight Process Improvements, January 8, 1999.
6. U. S. Nuclear Regulatory Commission, SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY 99-007), March 22, 1999.
7. U. S. Nuclear Regulatory Commission, SECY 01-0114, Results of the Initial Implementation of the New Reactor Oversight Process, June 25, 2001.
8. U. S. Nuclear Regulatory Commission, SECY 00-0049, Results of the Revised Reactor Oversight Process Pilot Program, February 24, 2000.
9. U. S. Nuclear Regulatory Commission Inspection Manual, Manual Chapter 0305, Operating Reactor Assessment Program, March 23, 2001.
10. NRC Inspection Manual, Manual Chapter 0609, Significance Determination Process, February 27, 2001.
11. Advisory Committee on Reactor Safeguards, NUREG-1635, Vol 4, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission, May 2001.

ADDITIONAL COMMENTS BY ACRS MEMBERS GEORGE E. APOSTOLAKIS, THOMAS S. KRESS, AND STEPHEN ROSEN

We agree with the recommendations and comments of our colleagues. The intent of our comments is to elaborate on the expert judgment process.

In any decisionmaking situation, the most important requirement is that the decisionmaker's judgments be consistent. This is particularly important for the ROP because the bases for the inputs to the action matrix are different.

One of the columns of the action matrix treats two white inputs and one yellow input (for one degraded cornerstone) as being equivalent. This means that the staff's judgment is that two white inputs signify a certain degradation in performance which is about the same as that corresponding to one yellow finding in the sense that the resulting regulatory response should be the same. For consistency in defining these color bands, one would have to address questions such as the following:

- Does the yellow band for the initiating event PI indicate a degradation in performance that is similar to that indicated by the yellow band for a mitigating system PI?
- Is the yellow band of a PI twice as important as its white band?
- Is a yellow finding from an SDP of equal significance as a finding that a PI is in its yellow band?

We appreciate that judgments such as "of equal significance" and "twice as important" are subjective. Our argument is that attempting to answer questions such as these removes a good deal of the subjectivity and, in fact, will be very helpful when the thresholds are determined. This argument acquires additional significance in the present case in which the action matrix does not represent the judgments of a single individual but those of the agency. In other words, communication among the experts who make these judgments would be enhanced.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 12, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDES ASSOCIATED WITH A PROPOSED REVISION TO 10 CFR 73.55, "REQUIREMENTS FOR PHYSICAL PROTECTION OF LICENSED ACTIVITIES IN NUCLEAR POWER REACTORS AGAINST RADIOLOGICAL SABOTAGE"

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, the Committee considered three draft regulatory guides (referenced) associated with a proposed revision to 10 CFR 73.55. The Committee plans to review the proposed final version of these regulatory guides following the reconciliation of public comments. The Committee has no objection to issuing these guides for public comment.

<u>References</u>:

1. Memorandum dated September 6, 2001, from Vonna Ordaz, NRR, to John T. Larkins, ACRS, Subject: Draft Regulatory Guides In Support of Proposed Rulemaking 10 CFR 73.55 w/atts:.

    a. U.S. Nuclear Regulatory Commission Draft Regulatory Guide 5011, "Standard Format and Content of Licensee Security Plans," issued September 17, 2001.
    b. U.S. Nuclear Regulatory Commission Draft Regulatory Guide 5012, "Standard Format and Content of Licensee Safeguards Contingency Plans," issued September 17, 2001.
    c. U.S. Nuclear Regulatory Commission Draft Regulatory Guide 5013, "Standard Format and Content of Licensee Training and Qualification Plans," issued September 17, 2001.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
B. Boger, NRR
G. Tracy, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 15, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE (GSI)-173A, "SPENT FUEL STORAGE POOL FOR OPERATING FACILITIES"

Dear Chairman Meserve:

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we met with representatives of the NRC staff to discuss the proposed resolution of GSI-173A, "Spent Fuel Storage Pool for Operating Facilities." We also had the benefit of the documents referenced.

RECOMMENDATION

The screening criteria used in the staff's proposed resolution of GSI-173A are appropriate to resolve the issue.

DISCUSSION

In its proposed resolution of GSI-173A, the staff used screening criteria such that if the frequency of fuel uncovery were 10^{-6} to 10^{-5}/yr, further technical evaluation would be performed. If the frequency were less than 10^{-6}/yr, no further regulatory action would be considered.

In our report dated June 20, 2000, we raised the concern that the screening criteria, which were derived on the basis of steam-oxidation severe accident source terms and the Commission's Safety Goals, could be inappropriate for spent fuel pool (SFP) accidents because the source term for such accidents could be dominated by air oxidation of clad, which could substantially increase the release of fuel fines and ruthenium compared to steam oxidation releases. In that report, we recommended that the staff defer declaring its resolution of GSI-173A until it considered the findings of NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," in which the use of the screening criteria was reassessed in view of the potential air-oxidation source term.

The results of NUREG-1738 indicate that even if the large early release frequency were 10^{-5}/yr, the Commission's quantitative health objectives (QHOs) would still be met with the air-oxidation source term and that at 10^{-6}/yr the risk level would be at least one order of magnitude lower than the QHOs. Of course, the acceptability of these frequency criteria presumes the availability of an emergency response plan.

Our expectation is that the acceptable risk contribution from a subset of sequences should be on the order of one-tenth of the overall acceptance criteria. The above screening criteria appear to meet that expectation. In addition, the studies in NUREG-1353 and NUREG/CR-4982 estimate the actual frequency of spent fuel uncovery at operating reactors to be about 2×10^{-8}/yr. In view of these results, we concur with the staffs' position that GSI-173A be considered resolved.

Sincerely,

George E. Apostolakis
Chairman

References:
1. Letter dated June 11, 2001, from Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Resolution of Generic Safety Issue (GSI)-173A, "Spent Fuel Pool Cooling For Operating Plants."
2. Report dated June 20, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool For Operating Facilities."
3. U.S. Nuclear Regulatory Commission , NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001.
4. U.S. Nuclear Regulatory Commission, NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," published April 1989.
5. U.S. Nuclear Regulatory Commission, NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," published July 1987.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 15, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDE (DG)-1085, "STANDARD FORMAT AND CONTENT OF DECOMMISSIONING COST ESTIMATES FOR NUCLEAR POWER REACTORS," AND DRAFT NUREG-1713, "STANDARD REVIEW PLAN FOR DECOMMISSIONING COST ESTIMATES FOR NUCLEAR POWER REACTORS"

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, the Committee considered the draft Regulatory Guide DG-1085, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," and draft NUREG-1713, "Standard Review Plan For Decommissioning Cost Estimates for Nuclear Power Reactors." The Committee decided not to review these two documents and has no objection to issuing them for public comment.

References:

1. Draft Regulatory Guide DG-1085, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," dated August 2001.
2. Draft NUREG-1713, "Standard Review Plan For Decommissioning Cost Estimates for Nuclear Power Reactors," dated August 2001.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
D. Matthews, NRR
M. Ripley, NRR
M. Virgilio, NMSS
S. Treby, OGC
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 17, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555-0001

SUBJECT: DUANE ARNOLD ENERGY CENTER EXTENDED POWER UPRATE

Dear Chairman Meserve:

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we met with representatives of the NRC staff and the Nuclear Management Company to review the license amendment request for an increase in core thermal power for the Duane Arnold Energy Center (DAEC), pursuant to the General Electric Nuclear Energy Extended Power Uprate Program. Our subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during meetings held on June 12 and September 26-27, 2001. During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The DAEC application for the extended power uprate should be approved.

2. The Safety Evaluation Report (SER) should be revised to document adequately the technical resolution of the issues raised by the staff.

3. The staff should develop improved guidance on the detail to be provided in SERs and criteria for when independent assessments should be performed to complement its reviews of applicant submittals.

DISCUSSION

The Nuclear Management Company has requested an amendment to the DAEC operating license for a 15.3% increase over the plant's current operating power limit. Previously, the staff had approved a smaller power uprate. Consequently, the current application is for a power uprate of 20% over the originally licensed power. This is the largest power uprate ever considered for boiling water reactors (BWRs) in the United States. It is anticipated that many other licensees will request similarly large increases in the operating powers of BWRs. Consequently, we anticipate that staff review of the DAEC power uprate will be a template for future reviews and will set the expectations for many future power uprate applications.

A generic methodology for evaluating and justifying power uprates of up to 20% for BWRs has been developed by General Electric. This generic methodology has been approved by the staff. The DAEC application has adopted this methodology and, in fact, the NRC staff has used the methodology to guide its review of this power uprate application.

The power increase at DAEC will be achieved by increasing steam production, while holding liquid flow in the core, dome pressure and temperatures quite near current values. The increased steam production is achieved by "flattening" the core power profile, which involves increasing power generation in the outer regions of the core. There is an increase in feedwater flow to match the increased production of steam. Balance-of-plant modifications are required and will cause the DAEC power increase to be performed in two steps.

Many technical issues must be addressed in an application for power uprate. Of these, we consider five to be especially significant:

1. Susceptibility of the plant to ATWS (Anticipated Transients Without Scram)

2. ATWS recovery

3. Reduction in some of the times available for operator actions because of higher decay heat

4. Material degradation due to irradiation-assisted stress corrosion cracking (IASCC) of reactor internals and flow-assisted corrosion and fatigue of feedwater piping

5. Containment response to accident events involving higher decay heat levels

Our examinations of the staff's SER and Requests for Additional Information submitted by the staff to the applicant persuaded us that the staff had raised numerous, pertinent issues concerning the conformance of the power uprate to approved methodologies. Though we persuaded ourselves eventually that the DAEC power uprate could be accomplished safely, we found it difficult to obtain information on the technical resolution of the issues either in the staff's SER or in our meetings with the staff. An exception to this common difficulty was the resolution of issues concerning containment response to design-basis accident events. In this case, the staff provided us a report on comparisons of applicant analyses with analyses done using an independent computational tool.

We found it far more difficult to assure ourselves that the DAEC core is susceptible only to global power oscillations and does not need to consider local power oscillations. It was similarly difficult to assure that ATWS recovery methods were applicable to cores with flattened power profiles, that critical human actions had been identified with adequate independence by the staff, and that material degradation sensitivities had been adequately assessed.

Many of the challenges that we encountered in our review of the DAEC power uprate application could have been eased if the staff had improved guidance on the detail to be provided in SERs and developed criteria for when independent assessments should complement reviews of applicant submittals.

ACRS Members Mario Bonaca and F. Peter Ford did not participate in the Committee's review of this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated September 5, 2001, to John T. Larkins, ACRS, from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Draft Safety Evaluation for Duane Arnold Energy Center Extended Power Uprate (draft Predecisional report).
2. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1999 (Proprietary).
3. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," February 2000 (Proprietary)
4. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (Proprietary).
5. GE Nuclear Energy, Topical Report, NEDC-32992P, "ODYSY Application for Stability Licensing Calculations," October 2000 (Proprietary).
6. BWR Owners' Group Letter dated March 8, 1996, transmitting GE Nuclear Energy Licensing Topical Report, BWR Owners Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960-A, November 1995.
7. Report (draft final) from A. Cronenberg, ACRS, "Margin Reduction Estimates for Re-Licensed/Uprated Plants: Hatch Case Study," August 2001.
8. Response by Nuclear Management Company to ACRS Thermal-Hydraulic Phenomena Subcommittee question, undated, attached to October 3, 2001 Memorandum from P. Boehnert to ACRS Members.
9. U.S. Nuclear Regulatory Commission, Technical Evaluation Report, ISL-NSAD-NRC-01-001, "Duane Arnold Energy Center Extended Power Uprate Containment Analysis Audit Calculation," B. Gitnick, Information Systems Laboratory, Inc., July 2001.
10. Memorandum (undated) from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Responses to Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Questions Regarding Duane Arnold Energy Center Extended Power Uprate, attached to October 3, 2001, Memorandum from P. Boehnert, to ACRS Members (contains Proprietary information).
11. GE Nuclear Energy Licensing Topical Report, NEDC-32980P, Rev. 1, "Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate," April 2001 (Proprietary).
12. Nuclear Management Company Memorandums: Response to Request for Additional Information - Duane Arnold Energy Center Extended Power Uprate, dated April 9, March 23, April 16, April 16 (Proprietary), May 8 (Proprietary), May 10, May 11, May 11 (Proprietary), May 22, May 29 (Proprietary),and June 5, 2001.
13. Nuclear Management Company Memorandums: Response to Request for Additional Information - Extended Power Uprate, June 11, June 18, June 21, June 28, July 11, July

19, July 25, August 1 (proprietary), August 1(proprietary), August 10 (proprietary), August 16 (proprietary), and August 21, 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D C. 20555-0001

October 18, 2001

The Honorable Richard A. Meserve
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: NRC ACTION PLAN TO ADDRESS THE DIFFERING PROFESSIONAL OPINION ISSUES ON STEAM GENERATOR TUBE INTEGRITY

Dear Chairman Meserve:

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we reviewed the Action Plan developed by the NRC staff to address the differing professional opinion (DPO) issues on steam generator tube integrity. Our Subcommittee on Materials and Metallurgy had reviewed this Action Plan during its meeting on September 26, 2001. The purpose of our review was to determine whether the Action Plan adequately and appropriately responded to our recommendations included in NUREG-1740, "Voltage-Based Alternative Repair Criteria." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

CONCLUSION

The Action Plan appropriately and adequately responds to our recommendations concerning the DPO on Steam Generator Tube Integrity. In the discussion that follows, we provide detailed comments on elements of the Action Plan that might help to refine and improve the efforts.

BACKGROUND

In February 2001, we sent to the Executive Director for Operations (EDO) an assessment of the technical issues raised in the DPO concerning alternative repair criteria for steam generator tubes in pressurized water reactors. We concluded that alternative repair criteria were needed and that general features of the criteria and the condition monitoring program the staff had endorsed provide such criteria that could adequately protect public health and safety. We did find that the DPO raised substantive technical issues that merited consideration. We made several recommendations to the EDO. Some were directly applicable to the details of the alternative repair criteria. Others related to the general risk status of plants with degrading steam generator tubes regardless of whether these plants had adopted the alternative repair criteria. Of the various recommendations, seven deserve to be highlighted:

1. Evaluate the potential for propagating steam generator tube damage during rapid depressurization caused by a main steamline break.

2. Monitor performance in search for systematic deviations from the linear bound on the nonlinear processes of crack initiation and growth through steam generator tube walls.

3. Improve the database for the correlation of tube leakage with voltage used in the condition monitoring program for 7/8" tubes.

4. Improve the analysis and understanding of radioactive iodine behavior during design-basis accidents.

5. Use improved risk assessments to support analyses of exemptions from the alternative repair criteria.

6. Develop a description of the probability of detection of steam generator tube flaws that will accommodate improvements in instrumentation and techniques.

7. Develop better understanding of the behavior of degraded steam generator tubes under severe accident conditions.

We concluded that the research that would be required to address our recommendations could be prioritized and pursued within the existing NRC research program augmented as necessary with additional resources.

DISCUSSION

The Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research have jointly developed the Action Plan to address our recommendations contained in NUREG-1740. This Action Plan, which has been incorporated into NRR's existing Steam Generator Action Plan, consists of eleven major activities:

1. Investigate the effects of depressurization during a main steamline break on steam generator tube integrity.

2. Complete investigation of jet penetration of adjacent tubes.

3. Develop experimental information on source term attenuation on the secondary side of steam generators (ARTIST tests).

4. Develop a better understanding of steam generator tube behavior under severe accident conditions.

5. Develop improved methods of assessing risk associated with steam generator tubes under accident conditions.

6. Assess the technical basis for improving the probability of crack detection in steam generator tubes.

7. Assess the need for better leakage correlations as a function of voltage for 7/8" steam generator tubes.

8. Monitor the predictions of flaw growth for systematic deviations from expectations.

9. Assess the need for a more technically defensible treatment of radionuclide release to be used in safety analyses of design-basis events.

10. Develop a better mechanistic understanding of tube cracking processes

11. Resolve Generic Safety Issue 163, "Multiple Steam Generator Tube Leakage."

The Action Plan does, indeed, address our recommendations included in NUREG-1740. Time scales envisaged for the work are consistent with expectations we had when we formulated our recommendations. Although the proposed work has been well integrated with ongoing work on steam generator tube integrity, we do have comments on some of the specific activities of the Action Plan:

- The efforts to understand threats to tube integrity posed by depressurization during main steamline breaks (Item 1, above) depend heavily on computer code analyses. In the absence of defensible, conservative load predictions, there is a need to validate predictions of computer codes with experimental data on modes of motion of steam generator tube support plates and stresses that these motions place on steam generator tubes. As noted in NUREG-1740, extant experimental data on thermal hydraulics and forces on tube support plates during depressurization are suspect because of poor scaling of the experimental facilities.

- The NRC staff should actively participate in formulating and conducting the ARTIST tests to investigate decontamination on the secondary side of steam generators (Item 3, above) rather than simply waiting for the data from the tests to become available. Activities necessary to use and understand the data from the planned tests should be defined and included in the Action Plan.

- Plans for examining steam generator tube behavior under severe accident conditions (Item 4, above) are quite detailed. These plans should be augmented to include a detailed assessment of the understanding of loop-seal clearing and the subsequent behavior in the reactor coolant system.

- We are impressed by the progress that has been made in the modeling of mixing and flow in the steam generator input plenum using computational fluid dynamics (CFD) models. We believe that this work will serve as a good example of how the NRC can use CFD models to resolve complicated regulatory issues.

- The lack of a correlation between leakage and voltage for 7/8" tubes (Item 7, above) is perplexing, in view of the good correlation for the 3/4" tubes. The staff should investigate the reason for this.

- The proposed work in connection with developing a better understanding of radioactive iodine behavior under design-basis accident conditions (Item 9, above) suggests that the staff does not accept our recommendation. Certainly, the staff has not committed to develop further the existing, mechanistic models of the iodine spiking phenomenon.

- The effort to develop a mechanistic understanding of stress corrosion cracking and its relationship to voltage signals (Item 10, above), is very long-term in nature as would be expected. This work will be conducted under a continuing cooperative international research program on steam generator tube integrity.

Results of the research on the effects of jet impingement on adjacent tubes (Item 2, above) have shown that the probability of damage progression is low enough that it can be neglected in the accident analyses.

The Action Plan should provide valuable input on risk assessment, inspection processes, and periodicity to the evolving life management strategy for steam generators.

We look forward to continued interaction with the staff as results are obtained from its planned work to refine and improve the technical bases for the alternative repair criteria.

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

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated June 1, 2001, from William D. Travers, Executive Director for Operations, NRC, to George Apostolakis, Chairman, ACRS, Subject: Steam Generator Action Plan Revision to Address Differing Professional Opinion (DPO) on Steam Generator Tube Integrity Issues.
2. U.S. Nuclear Regulatory Commission, NUREG-1740, "Voltage-Based Alternative Repair Criteria," Advisory Committee on Reactor Safeguards, March 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 18, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: RESPONSE TO YOUR AUGUST 8, 2001, LETTER ON THE RISK-BASED PERFORMANCE INDICATORS: PHASE 1 REPORT

During the 485th and 486th meetings of the Advisory Committee on Reactor Safeguards, September 5-7 and October 4-6, 2001, the Committee considered your August 8, 2001 response to the ACRS letter dated June 19, 2001, concerning the risk-based performance indicators Phase 1 report. The Committee decided to address issues related to the use of performance indicators in its report dated October 12, 2001, on the reactor oversight process (ROP). The Committee plans to continue its review of the development of risk-based performance indicators and looks forward to working with the staff as progress is made in establishing and testing risk-informed methods to support the ROP. The Committee is interested in reviewing the SAPHIRE code and SPAR model development in more detail during future meetings.

<u>References</u>:

1. Letter dated August 8, 2001, from William D. Travers, Executive Director for Operations, NRC, to George E. Apostolakis, Chairman, Advisory Committee on Reactor Safeguards, Subject. Risk-Based Performance Indicators: Phase 1 Report.

2. Letter dated June 19, 2001, from George E. Apostolakis, Chairman, Advisory Committee on Reactor Safeguards to William D. Travers, Executive Director for Operations, NRC, Subject: Risk-Based Performance Indicators: Phase 1 Report.

3. Report dated October 12, 2001, from George E. Apostolakis, Chairman, Advisory Committee on Reactor Safeguards, to Richard A. Meserve, Chairman, NRC, Subject: The Reactor Oversight Process (ROP).

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES
S. Newberry, RES
P. Baranowsky, RES
S. Mays, RES
S. Collins, NRR
M. Johnson, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 23, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: EPRI REPORT ON RESOLUTION OF NRC GENERIC LETTER 96-06 WATERHAMMER ISSUES

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we completed our review of the Electric Power Research Institute (EPRI) proposed approach to resolution of issues associated with waterhammer events occurring in low-pressure containment cooling systems in pressurized water reactors (PWRs), pursuant to the requirements specified in NRC Generic Letter (GL) 96-06. We also discussed this matter with representatives of the NRC staff and EPRI during our 485th meeting on September 5-7, 2001. In addition, our subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during meetings held on November 17, 1999, January 16-17, 2001, and August 22-23, 2001. During our review, we had the benefit of the documents referenced.

RECOMMENDATION

The application of the proposed EPRI methodology should not be approved until there is a better demonstration that it provides results that are bounding for realistic plant configurations and scenarios.

DISCUSSION

Fan cooler units (FCUs) are installed in PWRs to cool the containment during normal operation and, for some designs, to cope with the design-basis loss-of-coolant accident (LOCA) or main steamline break (MSLB) event simultaneous with a loss of offsite power (LOOP). In this design-basis event, the cooling water will stop circulating through the FCU and the fans driving containment air and steam across the FCU heat transfer surfaces will coast down. These conditions will last at least 30 seconds until the emergency diesel generators can start and the load sequencing will restart the service water pumps. This 30-second window is enough time for the water in the FCUs to boil and drain to create steam-and-air-filled void regions. When the FCU pumps return to power, a "slug" of water entering the voids creates conditions for a possible waterhammer event that could break the system piping, thus potentially causing loss of the cooling function, containment flooding, and creation of a containment bypass path.

The NRC issued GL-96-06, in part, to address the above concerns. The GL referenced NUREG/CR-5220, which provides a conservative approach for evaluating waterhammer events. EPRI and a number of utilities elected to pursue a less conservative approach than that specified in this report.

The EPRI methodology includes an analytical model of the closure of an air and steam pocket, which could cushion the impact of the incoming water slug, thus spreading out the pressure spike and reducing its maximum amplitude. The EPRI model hypothesizes that the pocket contains dissolved air that is released during boiling, and steam that is left uncondensed before the waterhammer event occurs.

The determination of the two major parameters (air release fraction and steam condensation on the water/void interfaces) that affect the reduction in severity of the waterhammer, was done by EPRI in scaled experiments that were intended to represent conditions in real FCUs.

To determine the air release fraction, EPRI conducted two series of experiments with two different configurations of the test apparatus. In the first configuration, water in a simulated FCU heat exchanger tube was allowed to drain into a header as boiling occurred in the tube due to heat addition from steam external to the tube. The difference between the oxygen content of the drained water and the initial concentration before the experiment started was said to be an indication of the air released. EPRI ran a number of such tests at two different boiloff rates and recommended the lower mean value of the results as the air release fraction under these conditions.

To simulate conditions that would exist in some parts of the FCU, the second configuration had the header release path filled with air-saturated water to a height of two-feet above the heat exchanger tube drain point, thus inhibiting drainage. In these experiments, the steam and released air bubbled up through the two-feet of water before being released. The dissolved oxygen content of the two-foot column before and after the 30-second test period was considered to be an indication of the air-release fraction. EPRI recommended using the lower mean value from the two sets of fractions of air released.

EPRI also experimentally determined the condensation rates of steam during column closure events in a two-inch diameter pipe. The real FCU waterhammer events are believed to generally take place in piping that is 10-inches to 16-inches in diameter. The results, therefore, must be scaled up to represent plant FCUs. EPRI developed a scaling model to do so.

Our discussion focused on the prototypicality of these experiments, the adequacy of the scaling model, and the appropriateness of the condensation and air-release models (see the attached discussion by ACRS Member Graham Wallis). We find that EPRI's conceptual model is oversimplified, and we are uncertain how it can be applied to plant-specific scenarios and configurations.

The combination of a LOCA and an independent LOOP is a very low probability event (some estimates have placed the frequency at a value as low as 10^{-9}/yr). A risk-informed view could conclude that such an event is not risk significant. This LOCA/LOOP event, however, has been defined as a design-basis accident, and coping with it must be considered as a compliance issue at this time. We would, however, support efforts to use risk-informed approaches to determine whether the FCU waterhammer requirements associated with the LOOP/LOCA event should be modified.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," September 30, 1996
2. EPRI Letter dated December 15, 2000, transmitting EPRI Report, TR-113594, Volumes 1 & 2, "Resolution of Generic Letter 96-06 Waterhammer Issues," December 2000 (Proprietary).
3. EPRI Letter dated July 10, 2001, to J. Tatum, U. S. Nuclear Regulatory Commission, Subject: Resolution of Generic Letter 96-06 Waterhammer Issues, EPRI Report TR-113594, Volumes 1&2, Revised Sections.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-5220, Vol. 1, "Diagnosis of Condensation-Induced Waterhammer," October 1988.

Attachment: Comments on EPRI Reports and Presentations on GL-96-06 (Fan Cooler Waterhammer Issues) by ACRS Member Graham Wallis, October 2, 2001.

COMMENTS ON EPRI REPORTS AND PRESENTATIONS ON GL 96-06
(FAN COOLER WATERHAMMER ISSUES)
Graham Wallis, October 2, 2001

In its earlier meetings with EPRI (November 17, 1999, January 16-17, 2001, and August 22-23, 2001), the Thermal-Hydraulic Phenomena Subcommittee raised three major issues: the justification for the air release model, the validity of the condensation model, and the relationship of these models to the actual events that have to be analyzed in a nuclear plant. EPRI responded to the first two of these at the Thermal-Hydraulic Phenomena Subcommittee meeting on August 22-23, 2001, and at the September 6-7, 2001 ACRS meeting.

I still have concerns over the third issue and how it influences the proper conclusions regarding the first two. EPRI has had little to say about plant scenarios. I am unconvinced that the work reported so far actually solves the real problem.

Layout of a Plant

The EPRI scenario is based on idealized sketches of a plant (Figures 2-1 and 2-2 of the Technical Basis Report). This is further reduced to a very simplified diagram in Figure 9-5 of the Report, which is also Figure 5-2 of the User's Manual. Analysis is based on the motion of a single slug compressing a single bubble in a single large pipe closed at the downstream end. All of the air released in the boiling process is assumed to be in this bubble.

The actual plant conditions are far from this conceptual model. Plants typically have three or four fan coolers. In the plants that I am familiar with, these are connected in parallel to ring headers, running around the periphery near the inner wall of containment, that supply water to the coolers and remove it from them. These ring mains also supply service water to other parts of the plant, therefore they are much bigger than is needed to carry the flows to the coolers themselves.

The coolers may each have similar designs, but the piping that connects them to the ring headers is likely to differ between coolers because of the need to avoid other structures in the vicinity. This connecting piping contains various bends and pipes of several orientations, which may include a vertical U-bend. The coolers are not necessarily installed in a symmetrical pattern with respect to the main supply lines or exhaust lines to the ring headers, which are represented by the vertical branches inside containment in EPRI's sketches.

The fan cooler module sketched in Figure 2-3 of the Technical Basis Report has 48 tubes per pass, each consisting of three 112-inch lengths of 5/8-inch tubing. The total fluid volume in this, assuming that 5/8 is the internal diameter of the tubes, is $3 \times 48 \times 112/1728 \times \pi/4 \times (5/8)^2 = 2.86$ cubic feet. According to the sketch, there are six modules stacked above each other. This would lead to an overall height of about 20 feet, which my colleagues who are familiar with plants tell me is too big. Accepting EPRI's figure, the total volume of the cooler is $6 \times 2.86 = 17.2$ cubic feet.

The ring mains have diameters as large as 16 inches. If the nominal value of the diameter is taken as close enough to the actual physical value, the volume of one of these is $\pi/4 \times (16/12)^2$ = 1.4 cubic feet per foot length. Since these mains are perhaps 200 feet long, the total volume of water that is initially in them is large compared with the volume of water in the fan coolers. This water is initially at around the containment temperature of perhaps 100° F and it is not significantly heated by the LOCA atmosphere during the 30 seconds of fan cooler voiding, because the main pipes are large and insulated. Some plants use chilled water, which is initially at or about 50^0 F

LOOP Alone

EPRI does not explain the details of how voids form in a LOOP scenario, but some events can be inferred. Looking at the LOOP discussion on pages 5-2 to 5-5 of their report, we see that plants with open-loop cooling water systems and either top- or bottom-draining FCUs experienced waterhammers during a LOOP. Apparently, this has not happened in plants that have closed-loop cooling water systems.

In a LOOP without a LOCA, the water is not heated and voids form at close to zero pressure. The voids contain a mixture of air and steam with such low density that they might be considered to be empty for some purposes.

We are told that waterhammer occurred in the main piping in both systems. Now, if the void were only in the top of the cooler with a bottom drain, turning on the pump would cause closure in the cooler unless the void actually extended to the main piping. So, I infer that the draining of the main piping, when there is almost a complete vacuum in the cooler, is so rapid that the voids extend into the main pipes before the pumps come on. Condensation does not occur because the temperature of 100° F is the same as the saturation temperature corresponding to the pressure (about 1 psia).

The key point is that in the time before the pumps come on, with an open-loop cooling water system, there is enough draining to the outside world to create significant voids in the main pipes as well as in the cooler, even when there is almost a complete vacuum sucking the water back. These voids are not going to look like EPRI's with vertical ends. They are more likely to be composed of long, stratified regions, with perhaps a bubble-like "nose" at the ends spreading along the pipe, as well as some regions of dispersed smaller bubbles.

I don't know what the assumptions are behind the typical plant's analysis of a LOOP without LOCA. If the void formation in the cooler is homogeneous, then any fluid leaving the cooler will carry air and steam with it, as I will hypothesize for the LOCA. If there is significant phase separation, then the cooler alone must void to some extent before voids reach the mains, depending on the details of the piping.

LOOP-LOCA

In a LOOP-LOCA event, the pumps circulating water to the fan coolers run down and stop. As the containment heats up, heat transfer to the fan coolers is very effective as the fans are still coasting down. Eventually, the water in them reaches saturation temperature corresponding to the local pressure and is ready to boil. Probably, the elevation differences from top to bottom of the cooler are small enough, especially if the height estimated by my colleagues who are familiar with these devices is less than half that shown by EPRI, that the conditions in the cooler may be taken as approximately uniform.

The water cannot boil unless room is made for the bubbles that are formed. Otherwise the pressure would rise, but no significant vaporization would occur. Voids open up when water flows out of the open ends of the piping system in an open-loop arrangement or when the level rises in the head tank in a closed-loop system. If there is a check valve at the pump in an open-loop system, water must flow out of the discharge, which will not be closed as shown in Figure 9-5 of the EPRI Report and in Figure 5-2 of the User's Manual.

The engineer faced with analyzing what happens next has to perform a transient dynamic analysis of the motion of the water, given a driving pressure (saturation?) in the cooler, the resistance and inertia of the fluid in all the flow paths, and the exhaust or head pressure. This may also involve some heat transfer calculations for the cooler, but probably this is not the limiting process, at least at the start.

Growth of the voids in the coolers leads to ejection of a steam/air/water mixture into the ring headers which are initially filled with water around the containment temperature of 100° F (or less if the water is chilled) and was not warmed up by the LOCA. The first fluid ejected is mostly water; later it is mostly steam. When this mixture emerges into the cold water, the steam tends to condense. This does not occur in a stagnant system. The water in these lines is itself set in motion to varying degrees in response to the flow out of the coolers. The flow out of one cooler displaces water past the other downstream coolers. The order of magnitude of the volumetric cocurrent flow of cold water is the same as that of the mixture leaving the cooler; therefore, new cold water is continually brought into contact with the fluid leaving the coolers.

Even if the flow in the main should be stagnant at one of the entry points, a large bubble of steam will spread by gravity along the top of the pipe in both directions at a speed given approximately by the standard formula for a large "Slug Flow" bubble, 0.5 times the square root of gD, or about 3.3 ft/s for a 16-inch pipe. As the bubble spreads out, this speed reduces roughly in proportion to the square root of the height of the vapor in the stratified region behind the bubble nose. This continually exposes new cold water at the ends of the bubble, and also stirs up the hot and cold water mixture. As long as the pressure is maintained above the saturation pressure corresponding to some effective value of the water temperature in the main, most of the steam entering the ring main(s) will condense leaving bubbles of air spread out along the top of the pipe. Estimates of the rate and extent of this condensation will be given later.

Eventually, the coolers may dry out, stopping the production of steam. If boiling is sufficiently vigorous, as it is described to be by EPRI, then the flow in the cooler may be approximated as homogeneous, with the air, steam, and water moving together at the same velocity. For this case, if the heat transfer and pressure conditions are approximately uniform throughout the cooler, it is possible to calculate the amount of steam that will have been driven into the ring headers by the time the cooler goes dry (see the Appendix to this discussion). The same result is predicted if the fluids in the cooler are well-mixed. For a pressure of one atmosphere in the cooler (closed system), the volume of steam driven out is 6.4 times the volume of the cooler, V_c, and the volume of water driven out is roughly the same as the volume of the cooler. For a pressure of 8 psia in the cooler (open-loop system, corresponding closely to a pressure of one-half atmosphere), the amount of vapor driven out is about 7 times the volume of the cooler.

This may seem like a lot of steam, but it has little mass. The mass is $6.4V_c/v_g$ in the first case. If it is condensed by water initially at 100° F the capacity of the cold water to condense steam may be evaluated from an energy balance in which the water is heated to the saturation temperature in the limiting case. The volume of water that it takes to condense all the steam is

$$V_w = 6.4V_c\, v_f/v_g\, h_{fg}/\,(212 - 100)\, C_p =$$
$$6.4\ (.01672/26.8) \times 970.4/(112 \times 1) = 0.0346V_c \text{ cubic feet.}$$

Therefore, it takes a volume of water equal to about 3.5% of the volume of the cooler to condense all the steam that is ejected from the cooler if these events occur at around atmospheric pressure. Using the cooler volume of 17.2 cubic feet calculated earlier, this water volume is close to 0.6 cubic feet or the amount of cold water in about a five-inch length of the ring main. The amount of cold water available in the ring mains is two orders of magnitude larger than this.

For the case of the open-loop cooling systems, EPRI states on page 6-5 that a typical pressure during voiding is 15 inches of mercury. For convenience in using steam tables, I'll assume that this corresponds to 8 psia.

Using the result in the Appendix, the amount of steam ejected from a cooler that is boiled dry homogeneously at 8 psia is 17.2 cu.ft. x 7 / 47.35 cu.ft./lb. = 2.54lb.

The amount of heat transfer it takes to condense this steam is 2.54lb x 988.5 BTU/lb = 2514 BTU. This corresponds to heating 30 pounds of water from 100° F to the saturation temperature of 182.8° F. This is about the amount of water contained in a 4.25-inch length of 16-inch pipe.

The volume of fluid ejected in boiling the cooler dry homogeneously at 8 psia is made up of seven times the cooler volume of steam and one cooler volume of water. Thus, the total volume ejected is 17.2 x 8 = 137.6 ft^3. The velocity of this if it filled a 16-inch pipe and occurred uniformly over 20 seconds is around 5 ft/s. In the smaller exhaust pipe from the cooler, this velocity will be bigger and will stir up the water in the main considerably, thereby enhancing mixing and condensation.

If the heat transfer coefficient is 72,000 BTU/hr.ft^2 F and it acts over an area of twice the main pipe cross-sectional area, then the time taken to condense this steam with a temperature difference of 82.8° F is

$$2514 \text{ BTU/lb} \times 3600 \text{ s/hr} / 72000 \text{ BTU/hr.ft}^2\text{F} / 82.8°\text{F} / 2 / 1.4 \text{ ft}^2 = 0.54 \text{ s.}$$

If the actual effective temperature difference is only 10° F, then the time is 4.5 s.

These results suggest that there is plenty of time for essentially complete condensation to occur in the mains during the voiding of the coolers, which takes around 20 seconds.

One may argue how efficient the mixing is in the ring main, but it is very likely that with so little cold water required, the steam will indeed all be condensed if the pressure is indeed 8 psia. The hot water, equal to about the cooler volume, that also enters the ring main is dispersed and mixed and is unlikely to inhibit condensation. Moreover, it is mostly ejected at the start of the process, while almost pure steam emerges over the latter stages of the discharge and it will be condensed very rapidly.

Should the cooler boil dry, the steam that is left in the cooler at the end of the process has a concentration of air equal to the initial concentration of air in the water. This is because the flow was homogeneous (the same result is obtained if it is considered to be well mixed). The mass of water corresponding to this volume is V_c/v_g whereas the initial mass of water was V_c/v_f. The ratio of these masses is 1600 at atmospheric pressure. Therefore, an estimate that one half of the air initially in the cooler is in the remaining steam bubble would be off by a factor of about 800. The air that was emitted from the water is almost all in the ring main. This illustrates that one cannot simply measure the amount of air removed from a water sample by boiling in a simple laboratory test, one has to work out where it goes in an actual plant. The discussion on pages 5-5 and 5-6 about "the void" does not take into consideration that there are different voids produced at different times. They have different histories and contain different amounts of air.

When the pumps are turned on, water flows into the fan coolers and will rapidly condense the vapor that is there. This is particularly true in the open-system with a check valve at the pump. Since in this case, no fluid comes from the cooler to the upstream ring main during the vaporization phase, the water that enters has not been preheated at all by steam or hot water. It meets steam with almost no air in it. If there is still boiling occurring in a fan cooler that has not fully drained by the time the pumps come on, then a balance must be computed between condensation in one part and boiling in another. There will also be somewhat more air in the steam in this case. I suspect that the condensation will overwhelm any vapor formation because the water flow rate is high and dispersed over many tubes.

In a **closed cooling water system**, or an open system with insufficient downstream hydrostatic suction to support a vacuum, some water from the downstream ring main may be sucked back into some or all coolers in response to rapid condensation in the cooler and the resultant depressurization. A dynamic system analysis has to be performed to decide the direction of flow of this water. Because the air bubbles and warm water have earlier been convected away, this water may be representative of the original cold water in the main.

If water enters a cooler from both ends, with the predominant flow at the entry because the pump head is far greater than the small difference between the vacuum in the cooler and the gravitational head downstream, the final column closure will occur in the cooler. Probably, the last vestiges of steam will be in the downstream header. They will be condensed by multiple jets emerging from the tubes, and will not produce a coordinated waterhammer, though there may be some bangs as individual small bubbles condense.

It is possible that conditions may favor the emergence of some steam into the ring main during the filling of the cooler. If the pressure is maintained at around the 8 psia value used by EPRI, this steam will be condensed on arrival in the cold water environment and will not produce a large bubble in the ring main. There may be some waterhammers in the piping leading from the cooler to the main, perhaps in the U-bends or other places favorable to the trapping of a steam bubble. This is a plant-specific question that is not answered by the EPRI work. Since there have probably been waterhammers in the cooler system during startup and LOOP testing, this event has probably already been shown not to damage the plant.

Even in the very unlikely event that steam survives to make a significant bubble in the ring main, there will be four of these bubbles, not one, and their interaction may tend to reduce the intensity of any waterhammer.

In plants with an **open loop cooling water system,** the rate of draining to the outside world plays a key role in the LOCA scenario. As the coolers boil, the pressure in the voiding region is kept up above the almost complete vacuum in the LOOP case. Therefore, the draining to the outside world is more rapid than in a LOOP alone. With this much water being lost, there must be a corresponding volume of voids formed and it will be larger than in the LOOP case.

In the LOOP event, the pressure is around 1 psia and voids immediately enter the mains if the voiding in the coolers is homogeneous. There, they will not condense and may coalesce to form large bubbles. On the other hand, though voids form more rapidly in the LOOP-LOCA event because of boiling, the voids that enter the mains are removed by condensation as long as the pressure is maintained sufficiently above the saturation pressure corresponding to the temperature in the mains. No waterhammer will occur in the mains unless there is a mechanism for voids to persist there.

Voids will form in the coolers but, as argued above, they will not extend to the main pipes if the steam formed (at 8 psia?) is being consumed almost at once by condensation on the cold water in the mains. What has to happen for voids to persist in the mains is for the pressure to drop to the point where condensation is no longer completely effective. In the extreme case, where approach to equilibrium is rapid, the energy balance is dominated by the heat capacity of the cold water in the mains. The pressure will drop to around the saturation temperature at 100° F, or 1psia, as in the LOOP. There will be a kind of "ejector" sucking steam out of the cooler. The flow rates and pressures at various points have to be calculated from an integrated analysis of the boiling, condensation, flow, and mixing phenomena. EPRI ran their tests at about one-half an atmosphere on the basis of the assertion that this was the pressure in the coolers during a LOCA-LOOP. This assertion must be the result of some analysis that is not described in their report and which needs to be explained.

A possible scenario for waterhammer in the main piping of a plant with an open-loop cooling water system is as follows. The coolers boil and voids form. Steam is ejected into the mains and it condenses, leaving air bubbles and warm water in parts of the mains. Because the mains are draining faster than in a LOOP, the voids in the fan cooler grow faster, making more steam. The pressure in the fan coolers has to be calculated by considering interaction between the rate of boiling, the rate of condensation, and the rate of draining of the main pipes to the outside world. This essential role of the rate and amount of draining to the outside world in an open loop system is missing from EPRI's report. Indeed, Figure 9-5 in the report, that shows the downstream end of the main pipe to be closed, is misleading as it precludes what is actually the governing process in void growth.

After a while, and especially if the fan coolers become dry, the draining to the outside world may drive the pressure low enough that condensation is no longer so effective in the mains. The pressure may also drop to the point where water in the mains that has been heated by condensation and by mixing with hot water ejected from the fan coolers may flash to form steam. At that time, the steam in the coolers will contain very little air, but the steam formed in the mains will probably mix with the air that was left there by earlier condensation. The point is that the histories of local pressures, temperatures, and air and steam volume fractions have to be calculated from some plant-specific system mode. There may be voids in several locations, each containing different amounts of air. Because there are several coolers and the flow and heat transfer histories differ in the mains downstream of each, it is not clear that subsequent events can be modeled conservatively by assuming a single bubble collapse in a single location. There may be a succession of column closures, with perhaps some of them involving steam with little air in it.

In contrast to the above, it seems unlikely that either LOOP or LOCA can lead to waterhammer in a closed-loop system because the pressure is maintained above atmospheric by the head tank (assuming it is above the coolers). I note that EPRI only mentions LOOP waterhammers in open-loop systems (pages 5-2 to 5-5). Steam voids are unlikely to get into the mains without condensing there during a LOOP-LOCA if there is a head tank because the key mechanism of a vacuum pulled by the outside world is absent and draining of the mains does not occur. There is plenty of cold water to condense the steam at the prevailing pressure and direct contact condensation should ensure that it occurs rapidly enough that little steam can survive in the main pipes.

There are still questions about the way in which the cooler voids. Does some sort of phase separation occur? How does it differ in the LOOP and LOOP-LOCA cases? Can it be assumed that the cooler is a single uniform node, or should it be represented by several nodes stacked vertically and horizontally? Boiling tends to make things homogeneous; does "void formation" at 1 psia do the same? I expect that void formation occurs throughout the fluid, as in a depressurized champagne bottle.

EPRI has not provided an analysis of the phenomena that occur in an actual plant. The very simplified problem that was addressed seems to have been formulated by a group of experts making estimates in setting up a Phenomena Identification and Ranking Table (PIRT). Perhaps they did have access to plant-specific analyses, but they haven't shown them to us nor explained the assumptions behind them. We and the staff deserve to see some plant analyses and may need to understand their bases in order to be able to tell if and how the proposed generic solution is relevant or leads to conservative predictions.

A code such as RELAP may have trouble modeling these events because it does not have a good representation of direct contact condensation, the mixing and stratification processes in the main pipes, nor the phase separation that may occur in the coolers.

It is reasonable to argue that waterhammer in a LOOP event would be expected to be more severe than that in a LOOP/LOCA because there is essentially no cushioning in the former case. However, this argument depends on a suitable demonstration that the other key parameters, such as closure velocities and void locations, are sufficiently similar for the two events in a particular plant.

APPENDIX

Let M be the mass of fluid in the cooler at a steam quality x. The volume of the cooler is related to its contents by

$$V_c = M\,(xv_g + (1\text{-}x)\,v_f) \qquad (1)$$

Let a mass –dM, containing a mass dM_g of steam leave the cooler. Then

$$-x\,dM = dM_g \qquad (2)$$

Eliminating x from (1) and (2) we get

$$dM_g = -dM/v_{fg}\,(V_c/M - v_f) \qquad (3)$$

Integrating from M_1 at the start to M_2 at the end of the process, the mass of steam ejected is

$$M_g = v_f/v_{fg}\,(M_2 - M_1) - V_c/v_{fg}\,\ln(M_2/M_1) \qquad (4)$$

If the cooler is initially full of water and ends up full of steam,

$$V_c = M_1 v_f = M_2 v_g \qquad (5)$$

Tidying up between (4) and (5),

$$M_g = M_1\,(v_f/v_{fg})\,(\ln\,(v_g/v_f) - 1) \qquad (6)$$

Or, in terms of volumes,

$$V_g = V_c\,(v_g/v_{fg})\,((\ln(v_g/v_f) - 1) \qquad (7)$$

Since v_g is very close to v_{fg} at low pressures, the first factor is about unity.

At one atmosphere, v_g= 26.8 cu.ft/lb and v_f= 0.01672 cu.ft /lb. Then (7) reduces to

$$V_g = V_c\,(7.4 - 1) = 6.4\,V_c.$$

Equation (1) is valid if the conditions of each fluid element are the same and they are each homogeneous (no mixing). It is equally valid if the contents of the cooler are well-mixed. The physics and phase distributions do not change from one case to the other, the conditions being uniform in each case.

The same result can be deduced by analyzing the special case of voiding at constant pressure of a long pipe heated with a uniform heat flux. It is not necessary that the heat flux be constant with time. In this case, the actual void history of each fluid particle is also predicted, perhaps lending more credibility to the analysis until it is realized that the actual geometry does not matter.

The nomenclature used in the above equations is as follows:

C_p	specific heat of water
D	pipe diameter
g	acceleration due to gravity
h_{fg}	latent heat
M	mass
v_f	specific volume of water
v_g	specific volume of steam
v_{fg}	change of specific volume in evaporation
V_c	volume of cooler
V_w	volume of water
x	steam quality

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 23, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDES ASSOCIATED WITH A PROPOSED REVISION TO 10 CFR 50.55a, "CODES AND STANDARDS"

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, the Committee considered four draft regulatory guides associated with a proposed revision to 10 CFR 50.55a. The Committee plans to review the proposed final version of these regulatory guides following the reconciliation of public comments. The Committee has no objection to issuing these guides for public comment.

References:
1. Memorandum dated September 27, 2001, from Michael Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: ACRS Review of Draft ASME Code Case Regulatory Guides.
2. U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1090 (Proposed Revision 32 to Regulatory Guide 1.84), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," received September 27, 2001.
3. U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1091 (Proposed Revision 13 to Regulatory Guide 1.147), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," received September 27, 2001.
4. U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1089, "Operation and Maintenance Code Case Acceptability, ASME OM Code," received September 27, 2001.
5. U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1112, "ASME Code Cases Not Approved for Use," received September 27, 2001.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
W. Norris, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 13, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: CORE POWER UPRATES FOR DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS

During the 487th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 2001, the Committee met with representatives of the NRC staff and the Exelon Generation Company to review the license amendment requests for an increase in core thermal power for the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, pursuant to the General Electric Nuclear Energy Extended Power Uprate Program. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during a meeting held on October 25-26, 2001.

The Committee cannot complete its review of this matter until the NRC staff resolves the open issues, particularly the issue of the need for conducting large transient tests. The Committee would like to discuss the resolution of these issues with the staff during its December 2001 meeting and anticipates concluding its review at that time.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
B. Sheron, NRR
J. Zwolinski, NRR
S. Bajwa, NRR
M. Shuaibi, NRR
L. Rossbach, NRR
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 14, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: For John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: UPDATE RULEMAKING FOR 10 CFR PART 52

During the 487th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 2001, the Committee considered the draft rule language for the proposed update to 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." The Committee plans to review the proposed final version of 10 CFR Part 52 following the reconciliation of public comments.

References:

1. Memorandum dated September 27, 2001, from James E. Lyons, Office of Nuclear Reactor Regulation, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Update Rulemaking for 10 CFR Part 52.
2. Letter dated November 8, 2001, from Ron Simard, Nuclear Energy Institute, to Annette L. Vietti-Cook, Secretary, NRC, Subject: Comments on Petitions for Rulemaking, Docket Numbers PRM-52-1 and PRM-52-2.
3. Letters dated July 18, 2001, from Robert W. Bishop, Nuclear Energy Institute, to Annette L. Vietti-Cook, NRC, Subject: Petitions for Rulemaking for 10 CFR Part 52.
4. Memorandum dated August 2, 2001, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Risk-Informing Special Treatment Requirements.
5. Memorandum dated February 13, 2001, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Staff Readiness for New Nuclear Plant Construction and the Pebble Bed Reactor.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
J. Lyons, NRR
J. Wilson, NRR
A. Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 16, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

During the 487th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 2001, we completed our review of the Southern Nuclear Operating Company's (SNC's) application for license renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2, and the related final Safety Evaluation Report (SER). We issued an interim letter concerning this application and the SER with open items on April 16, 2001, and our Plant License Renewal Subcommittee held discussions with representatives of the staff and SNC on October 25, 2001. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The SNC application for renewal of the operating licenses for Hatch, Units 1 and 2, should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that Hatch, Units 1 and 2, can be operated safely in accordance with their licensing bases for the period of extended operation without undue risk to the health and safety of the public.

3. The staff has performed a comprehensive review of SNC's application. The open items identified in the February 2001 draft SER have been resolved satisfactorily.

4. The SER clarifies staff positions on non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. These clarifications provide significant guidance that could prevent these issues from becoming open items in future applications. They should be incorporated into the generic license renewal guidance documents.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. SNC requested renewal of the operating licenses for Hatch, Units 1 and 2, for a period of 20 years beyond the current license terms, which expire on August 6, 2014, for Unit 1, and June 13, 2018, for Unit 2. The final SER documents the results of the staff's review of information submitted by SNC, including those commitments that were necessary to resolve open items identified by the staff in its February 2001 draft SER. The staff's review included the verification of the completeness of structures, systems, and components (SSCs) identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging effects associated with each passive long-lived component, and the verification of the adequacy of the aging management programs. The staff also conducted site inspections to verify the adequacy of the implementation of the methodology described in the application.

As noted in our April 16, 2001 interim letter, the SNC's approach to identifying SSCs that are within the scope of the License Renewal Rule is function-based, rather than the system-based approach used in previous applications. This approach was adequate, but made it difficult for the reviewers to ascertain which SSCs were in scope and which were not. The staff's review relied heavily on supporting documents located at the site and on requests for additional information. In addition, the staff performed a "walk-through" of the process for three systems that are within scope. On the basis of its extensive review, the staff identified some additional components that the applicant should have included within the scope of license renewal, and classified them as open items. These open items have been resolved by including the additional components in scope. We concur with the staff that the applicant has now properly identified SSCs requiring an aging management review.

Components brought into scope through the resolution of open items include non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. The inclusion of these components was contested in previous license renewal applications. The issue of seismic II-over-I piping is an open item in an application that is currently under review. The Hatch SER includes effective clarifications of why these components need to be included within scope. The guidance provided by these clarifications could prevent these issues from becoming open items in future applications. Consequently, these clarifications should be incorporated into the generic license renewal guidance documents.

SNC has conducted a comprehensive aging management review of SSCs that are within scope. Aging effects were identified on the basis of component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Topical reports developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) were also used to identify aging effects and to develop aging management programs that support the Hatch application. We reviewed a number of BWRVIP topical reports and commented on their effectiveness in supporting license renewal in our April 16, 2001 letter.

Appendix A to the Hatch application describes 17 existing programs, 5 modified programs, and 7 new programs that SNC has implemented to manage aging effects during the period of extended operation. The resolution of open items has resulted in added commitments to these programs, including a one-time inspection of plant service water piping in the diesel generator building and a one-time inspection of small-bore butt-welded stainless steel piping.

One of the added commitments resulting from resolution of open items involves periodic testing of fire-protection system sprinkler heads that are within the scope of license renewal. SNC had proposed a one-time test of such sprinkler heads at or before the start of the period of extended operation. The staff did not agree with the one-time test, because the design life (50 years) of the sprinkler heads does not cover the period of extended operation. As recommended by the staff, SNC has committed to perform the sprinkler head tests as specified in the National Fire Protection Association (NFPA) Standard 25, Section 2.3.3.1, "Sprinklers." The application of this Standard will result in periodic testing of the sprinkler heads at 10-year intervals, with the first test taking place during the third year of the renewal period. This program is acceptable because it confirms the effectiveness of the periodic inspections to which the sprinkler heads are subjected and ensures testing of the sprinkler heads early in the renewal period.

The staff requested that SNC perform a one-time inspection of the four buried emergency diesel generator (EDG) fuel oil storage tanks. SNC responded by performing visual inspections and ultrasonic testing of one of the four tanks. Ultrasonic testing of 144 locations along the lower shell of the tank indicated that there was no thinning of the wall. Visual inspections of the internal surface revealed very little corrosion. SNC and the staff concluded that the one-time inspection demonstrated that loss of material of the diesel fuel oil storage tanks was not an aging effect requiring management during the period of extended operation.

We also considered the possibility that the external coating of a tank could be damaged at some location during installation and result in localized fuel oil leakage. Such damage would be of concern during the current license term and, thus, would not be specific to the period of extended operation. The safety consequences would not be significant because the potential leakage would not cause substantial depletion of the fuel oil inventory before it would be detected. We concur with the staff's determination that loss of material of the diesel fuel oil storage tanks is not an aging effect requiring management during the period of extended operation.

Jet pump assemblies and fuel supports contain cast austenitic stainless steel (CASS) components that are within the scope of license renewal. These components may be exposed to neutron fluence levels that would make them susceptible to neutron irradiation embrittlement and loss of fracture toughness. Since neutron embrittlement becomes a concern when cracks are present in the components, the staff requested that SNC propose a one-time inspection of the jet pump assemblies and fuel supports to confirm that these CASS components have not experienced cracking. Following this request, the staff recognized that cracking of CASS components has not been observed to date. Furthermore, BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," requires inspections of jet pump assembly welds that are generally believed to be more susceptible to cracking than the CASS components and, therefore, provide a leading indicator for inspection of CASS components. SNC has committed to perform the weld inspection required by BWRVIP-41. In addition, the BWRVIP and the NRC's Office of Nuclear Regulatory Research plan to conduct confirmatory research to determine the effects of high levels of neutron fluence on BWR internals. SNC has committed to implement any requirements resulting from this research. Given the above, the staff concluded that the requested one-time inspection is not warranted at this time. We agree with the staff's conclusion.

Time-limited aging analyses (TLAA) have shown that neutron irradiation embrittlement during the extended period of operation will have no significant impact on the integrity of the Hatch reactor vessels. At the end of the renewal period, the vessels will still have margin over applicable regulatory limits. In order to monitor time-dependent parameters used in the TLAA, SNC plans to implement the provisions of the integrated surveillance program (ISP) described in BWRVIP-78, BWR integrated surveillance program plan, and BWRVIP-86, BWR integrated surveillance program implementation plan. Since these topical reports have not yet been approved by the staff, SNC committed to implement either a staff-approved ISP or a plant-specific program that meets specific staff requirements on periodic removal of capsules to monitor neutron fluence and the impact of irradiation on the reactor vessels. SNC committed to provide the staff with program details prior to the period of extended operation. The staff made this commitment a license condition.

The staff has performed a comprehensive review of SNC's application. The applicant and the staff have identified plausible aging effects associated with passive and long-lived components. Adequate programs have been established to manage the effects of aging so that Hatch, Units 1 and 2, can be operated safely in accordance with their current licensing bases for the period of extended operation.

Sincerely,

George E. Apostolakis

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued October 2001.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued February 2001.
3. Letter dated February 29, 2000, from H. L. Sumner, SNC, to the U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Application for Renewed Operating Licenses."
4. Letter dated April 16, 2001, from George E. Apostolakis, Chairman ACRS, to

William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter Related to the License Renewal of Edwin I. Hatch Nuclear Station, Units 1 and 2.

5. Topical Report BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," October 1997.
6. Topical Report BWRVIP-78, "BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material," December 1999.
7. Topical Report BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 10, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: FINAL PART 20 RULEMAKING ON REVISION OF THE SKIN DOSE LIMIT

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, the Committee considered the draft final rule on skin dose limit and decided not to review it. The Committee has no objection to issuing the final rule for industry use.

Reference:
Memorandum dated November 6, 2001, from D. B. Matthews, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request for Review of Final Part 20 Rulemaking on Revision of the Skin Dose Limit.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
B. Sheron, NRR
D. Matthews, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 10, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED CLOSEOUT OF GENERIC SAFETY ISSUE-172, "MULTIPLE SYSTEM RESPONSES PROGRAM"

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, the Committee considered the staff's proposed closeout of Generic Safety Issue (GSI)-172 and decided not to review it. The Committee has no objection to the staff's proposal to closeout GSI-172.

Reference:
Memorandum dated November 30, 2001, from Scott F. Newberry, RES, to John T. Larkins, ACRS, Subject: Proposed Closeout of Generic Safety Issue-172, "Multiple System Responses Program"

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
A. Thadani, RES
J. Ridgley, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 12, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: CORE POWER UPRATES FOR DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, we completed our review of the Exelon Generation Company (Exelon) license amendment requests for increases in core thermal power for the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, pursuant to the General Electric (GE) Nuclear Energy Extended Power Uprate Program. We had previously discussed this matter with representatives of the NRC staff and Exelon during our 487th meeting on November 7-9, 2001. Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting held on October 25-26, 2001. During our review, we had the benefit of the documents referenced.

RECOMMENDATION

We agree with the staff's recommendation that the Commission should issue license amendments that will permit increases in the licensed power levels of the Dresden and Quad Cities Nuclear Power Plants by 17% and 17.8%, respectively.

DISCUSSION

Exelon has requested amendments to the operating licenses of the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, for an increase in the operating power limits. Presently, the power limits are 2527 MWt for the Dresden Units and 2511 MWt for the Quad Cities Units. All four units would be uprated to 2957 MWt, which represents uprates of 17% and 17.8%, respectively. These four units employ the GE boiling water reactor (BWR/3) nuclear steam supply system and the Mark I containment design.

Exelon used the NRC-approved GE generic methodology (ELTR-1 and ELTR-2) for analyzing extended power uprates (EPUs). Precedents for EPU applications using these methods have been set by the Monticello (1998), Hatch (1998), and Duane Arnold (2001) nuclear power plant licensees. These uprate requests were approved by the Commission. These precedents have guided Exelon and the staff in the preparation and review of the current EPU applications.

The uprate applications for Dresden and Quad Cities are nearly identical, and are similar in concept to that for Duane Arnold. The increased power is achieved by use of a new fuel design; a sophisticated fuel management scheme is used to ensure that all regulatory limits for fuel behavior continue to be met. Both feedwater flow and steam flow rates are increased, but the dome pressure and overall core flow remain unchanged. The increased steam and feedwater flow rates require modifications to the steam dryers and the use of all feedwater and condensate pumps. Because the pressure and temperatures and the overall water inventory in the primary coolant system remain essentially unchanged, effects of the EPUs on design-basis accidents are relatively minor. There are slight increases in risk due primarily to the increased decay heat to be removed in loss-of-coolant accidents and to the shortened available time for operator action during events such as anticipated transients without scram.

The staff has determined that the proposed EPUs meet all regulatory criteria, and that the licensee has used approved predictive methods. In addition, the important materials degradation issues have been identified and adequate management programs are in place to monitor potential increases in degradation rates.

The GE Topical Report, ELTR-1, that supports the EPUs (Reference 6) includes the requirement that certain large transient tests be performed to confirm the effectiveness of the implemented plant modifications. GE has since reached the conclusion that these tests are not necessary for power uprates in which reactor steam dome pressure is not changed. The staff agrees with this conclusion for the Dresden and Quad Cities applications. Technical arguments to support this decision are documented in the Safety Evaluation (SE). We concur with the staff's conclusion for these plants.

In our report on the Duane Arnold Energy Center power uprate, we expressed concerns about the adequacy of the documentation of the staff's review, as reflected in its SE. We noted that many of the challenges that we encountered in that review would have been eased if the staff had provided more details concerning its review and the criteria used to reach conclusions in the SE. We have similar concerns about the Dresden and Quad Cities SEs. Frequently, the licensee's analysis is just summarized and the results of the staff's evaluation are represented by a statement that the analysis is "acceptable." These summary statements do not reflect the substantial effort that went into the staff's review, including audits conducted onsite and at vendor facilities. The depth of the review became more apparent during meetings at which the staff presented more details and responded to our questions. The staff's responses have given us sufficient assurance that an appropriate technical review has been performed and that the staff's conclusions are valid. Upgrading the SEs to better reflect the depth and breadth of the staff's engineering evaluations would provide the public a better sense of these activities and engender more confidence in the work of the NRC.

Although the depth and breadth of the staff's review of these uprates has been adequate, development of a Standard Review Plan Section would help ensure an adequate review of future power uprate applications. It would also clarify to both the public and licensees what is required for an application for power uprate to be found acceptable.

Dr. F. Peter Ford did not participate in the Committee's deliberations regarding this matter.

Additional comments by ACRS Member Stephen L. Rosen are provided below.

Sincerely,

George E. Apostolakis
Chairman

Additional Comments by ACRS Member Stephen L. Rosen

I concur with my colleagues that the requested EPU should be granted. However, I have the following additional comments.

The staff has accepted the applicant's arguments that it is unnecessary to conduct tests of the units' capability to successfully respond to a generator load rejection or a main steam isolation valve closure demand at the higher steam flows at EPU conditions. The applicant maintains that these tests were part of the units' initial startup testing and need not be repeated at EPU conditions.

The applicant's justification for this position includes the fact that reactor pressure is unchanged by the planned EPU and that unnecessary plant transients should be avoided. They also argue that no new plant equipment will be installed that could affect the units' demonstrated capability to respond to a generator load rejection or a main steam isolation valve closure demand.

The staff has also accepted the applicant's arguments that it is unnecessary to conduct integral testing of the new recirculation "run back" system. In this case, although new equipment will be added to the plant to rapidly reduce recirculation flow and reactor power to match feedwater flow in the event of a main feedwater or condensate pump trip, the applicant argues that overlapped simulated logic functional tests are sufficient.

The applicant's justification for this position is that a reactor scram will occur if "run back" is unsuccessful after a sudden feedwater flow reduction.

Generator load rejections, main steam isolation valve closure demands, and main feedwater or condensate pump trips are Anticipated Operational Occurrences with expected frequencies in the range of 1 per 1-10 years. In granting the applicant's EPU request without requiring performance of integral testing, the staff has relied on the applicant's "well established quality assurance programs including component and system level post-modification testing."

Since integral tests of the plants' response can reveal otherwise undetected latent flaws, these tests should be conducted to confirm that these programs have achieved the desired result.

I am not convinced by the applicant's arguments and the staff's conclusion that integral tests are not necessary. I believe approval of the EPU application should be conditioned on the successful completion of these tests shortly after reaching EPU conditions.

References:

1. Memorandum dated October 10, 2001, to John T. Larkins, ACRS, from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Draft Safety Evaluation for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, Extended Power Uprate (draft Predecisional report).
2. Draft Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Facility Operating License No. DPR-25, Exelon Generation Company, LLC, Dresden Nuclear Power Station, Units 2 and 3, received December 7, 2001.
3. GE Topical Report, NEDC-32962P, "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," dated December 2000 (Proprietary).
4. GE Topical Report, NEDC-32961P, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," dated December 2000 (Proprietary).
5. Letter dated December 27, 2000, from Commonwealth Edison Company, to U.S. NRC, Subject, Request for License Amendment for Power Uprate Operation.
6. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-1) February 1999 (Proprietary).
7. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-2) February 2000 (Proprietary).
8. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (ELTR-2) (Proprietary).
9. Exelon Generation Company Memorandums: Response to Requests for Additional Information Supporting License Amendment Requests to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station, dated August 9, August 14, August 31, August 31, September 5, September 5, September 14, September 19, September 25, September 26, September 27 (contains proprietary information), and September 27, 2001.
10. Exelon Generation Company Memorandums: Response to Requests for Additional Information Supporting License Amendment Requests to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station, dated August 8 (contains proprietary information), August 13 (contains proprietary information), August 13, August 14, and August 29, 2001.
11. Memorandum dated December 3, 2001, from P. Boehnert, ACRS, to ACRS Members, Subject: Dresden/Quad Cities Power Uprate - Exelon Response to ACRS Questions/Additional Information Regarding NRC Review of PRA.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 12, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RULEMAKING FOR RISK-INFORMED REVISIONS TO 10 CFR 50.44, "STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER-COOLED POWER REACTORS"

Dear Dr. Travers:

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, we held discussions with representatives of the NRC staff concerning the proposed rulemaking for risk-informed revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors." We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The proposed rulemaking for risk-informed revisions to 10 CFR 50.44 will provide more effective and efficient regulation to deal with combustible gases in containments.

2. The proposed specification for the combustible gas source term for boiling water reactor (BWR) Mark III and pressurized water reactor (PWR) ice condenser containments should be included in the draft regulatory guide (DG-1117) instead of being incorporated directly in the rule.

DISCUSSION

The severe accident research programs sponsored by the NRC during the 1980s and 1990s, the studies of severe accident risk (NUREG-1150), and the insights derived from Individual Plant Examinations (NUREG-1560) have led to an improved understanding of the behavior of combustible gas during reactor accidents. The staff provided a useful summary of the risk significance of combustible gases in Attachment 2 to SECY-00-0198, which we commented on in our report dated September 13, 2000. The severe accident studies have shown that control of combustible gases during design-basis accidents does not have significant impact on risk, but that controls are needed for beyond-design-basis accidents for some containment designs.

The proposed rule changes incorporate insights obtained from past NRC efforts. They retain requirements for high-point vents and for ensuring a mixed-containment atmosphere, inerting BWR Mark I and Mark II containments, and providing hydrogen control systems for Mark III and ice condenser containments. The proposed rule eliminates the design-basis loss-of-coolant accident hydrogen release and requirements for systems to mitigate such a release. It retains the requirement to monitor hydrogen in the containment atmosphere for all containment designs, but monitors are no longer classified as safety-related components. Also, the proposed rule would codify the existing regulatory practice of monitoring oxygen concentrations in containments with inerted atmospheres. In addition, the proposed rule includes a number of options offering performance-based and prescriptive alternatives.

In SECY-00-0198, the staff proposed to develop combustible gas source terms appropriate for different containment types and accident scenarios. In the proposed rule, the staff has chosen instead to continue the use of a prescriptive requirement for a source term equivalent to the hydrogen generated from metal-water reactions involving 75% of the fuel cladding surrounding the active fuel region. Because of the ongoing investigation of combustible gas source terms in support of resolution of Generic Safety Issue-189 on the potential need for regulatory enhancement to deal with station blackout sequence issues for BWR Mark III and PWR ice condenser containments, it is preferable not to prescribe the source term in the rule. A combustible gas source term should be included in the associated regulatory guide (DG-1117).

We would like to review the proposed final rule after reconciliation of public comments.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated November 20, 2001, from David B. Matthews, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Proposed Part 50 Rulemaking on Risk-Informed Revision of Combustible Gas Control (Predecisional)
2. Report dated September 13, 2000, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."
3. U.S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
4. U.S. Nuclear Regulatory Commission, NUREG-1560, Vols. 1-6, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997.
5. Memorandum dated January 19, 2001, from Anette L. Vietti-Cook, Secretary, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-00-0198 - Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).

6. Memorandum dated September 14, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommended Changes to 10 CFR 50.44 (Combustible Gas Control)."
7. Memorandum dated November 16, 2001, from Thomas L. King to Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, Subject: Generic Issue Management Control System Report -- Fourth Quarter FY 2001; Generic Safety Issue 189, "Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During a Severe Accident," identified May 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 14, 2001

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED STEAM GENERATOR PROGRAM GUIDELINES AND ASSOCIATED GENERIC LICENSE CHANGE PACKAGE

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, we reviewed the latest revision to Nuclear Energy Institute document, "Steam Generator Program Guidelines" (NEI 97-06), and the associated Steam Generator Program Generic License Change Package. Our Materials and Metallurgy Subcommittee reviewed these documents on November 29, 2001. During our reviews, we had the benefit of discussions with representatives of the staff and NEI. We also had the benefit of the documents referenced.

CONCLUSIONS

1. NEI 97-06 and the related Generic License Change Package describe a steam generator tube management program that is flexible enough to accommodate evolving technical knowledge and could provide an enforceable regulatory framework.

2. We concur with the staff's conclusion that there is a need for additional technical justification to support the industry's proposal to extend the inspection intervals for Alloy 600TT and 690TT tubing beyond that currently permitted by regulations.

BACKGROUND

In the early 1990s, the regulations required licensees to repair steam generator tubes having flaws deeper than 40 percent through-wall. Since the nondestructive examination (NDE) methods were unable to characterize the dimensions of a crack with sufficient accuracy and reproducibility, licensees repaired tubes with identified cracks. In 1995, the staff issued Generic Letter 95-05, which, in certain specific situations, allowed steam generator tubes with cracks to remain in service, in part, on the basis of data from voltage-based NDE methodologies.

Since that time, the staff and NEI have worked to develop a regulatory framework to ensure steam generator tube integrity. The staff considered developing a rule, but it failed to pass the regulatory analysis test.

In 1997, affected licensees committed to follow NEI 97-06 that defined performance criteria for structural integrity and leakage under accident and normal operating conditions. These criteria were implemented by tube integrity assessment guidelines defined in a series of evolving Electric Power Research Institute (EPRI) reports. In addition, NEI proposed an industry Steam Generator Program Generic License Change Package, which provided templates for licensees to amend plant-specific technical specifications.

The staff and NEI are now in general agreement concerning the content of NEI 97-06 and the degree of regulatory control offered by the Generic License Change Package. Of particular note is the fact that the program is adaptable and that the supporting technology is evolving. Investigations sponsored by the industry and the NRC staff are ongoing.

DISCUSSION

Our discussions with the staff and NEI addressed wide-ranging technical issues, including the following:

- Effectiveness of performance criteria for structural integrity and leakage rates in light of NDE uncertainties
- Qualification of NDE processes
- Adequacy of burst and leakage models
- Adequacy of the data to justify proposed inspection and condition monitoring intervals

An unresolved technical issue is the appropriate length of inspection intervals. The NEI position on inspection intervals is contained in draft revision 6 of the EPRI document, "PWR Steam Generator Examination Guidelines," in which NEI proposes that the intervals for Alloy 600TT and 690TT tubing be extended beyond the interval permitted by current regulations. Although the domestic operating experience justifies increasing the inspection intervals for these alloys, there have been reported incidents of cracking of Alloy 600TT tubes in foreign plants. These identified cracks have been discounted by industry due to different construction and operating conditions. However, there were no controlled data presented to explain the validity of this position and whether the differences in stress, environment, or material conditions, which govern cracking susceptibility, are sufficient to ensure an adequate resistance in domestic plants. There is a need to review the relevant global operating and laboratory databases.

The steam generator management program was originally intended to be primarily performance based. If the Generic License Change Package is approved, a greater degree of performance-based capability will have been achieved. Full implementation of a performance-based approach for determining plant-specific inspection intervals will depend on evolving developments in inspection techniques, and quantification of the stochastic aspects of stress corrosion cracking.

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

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated February 7, 2001, from David Modeen, Nuclear Energy Institute (NEI), to Samuel J. Collins, Office of Nuclear Reactor Regulation (NRR), NRC, Subject: NEI 97-06, "Steam Generator Program Guidelines," Revision 1.
2. Letter dated December 11, 2000, from David Modeen, NEI, to Samuel J. Collins, NRR, Subject: Revised Industry Steam Generator Program Generic License Change Package.
3. Memorandum dated September 18, 2001, from Edmund J. Sullivan, Division of Engineering, NRR, to William Bateman, Division of Engineering, NRR, Subject: NRC Staff Comments on Steam Generator Inspection Intervals.
4. Memorandum dated September 21, 2001, from Maitri Banerjee, Division of Licensing Project Management, NRR, to Edmund J. Sullivan, Division of Engineering, NRR, Subject: "Summary of August 29, 2001, Public Meeting With the Nuclear Energy Institute Regarding NEI 97-06."
5. U. S. Nuclear Regulatory Commission Generic Letter 95-05 -- Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, August 3, 1995.
6. Letter dated May 15, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Final Generic Letter 95-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 14, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED RULEMAKING ON SUBMISSION OF FINANCIAL INFORMATION REQUIREMENTS FOR APPLICATIONS TO RENEW OR EXTEND THE TERM OF AN OPERATING LICENSE FOR POWER REACTORS (10 CFR §50.33(f)(2))

During the 488th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2001, the Committee considered the proposed rulemaking and decided not to review it.

The Committee has no objections to issuing this proposed rule for public comment.

Reference:
Memorandum dated November 15, 2001, from David B. Matthews, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, SUBJECT: Request for Review of Proposed Rule Part 50 Rulemaking on Submission of Financial Information Requirements for Applications to Renew or Extend the Term of an Operating License for Power Reactors.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
D. Matthews, NRR
C. Carpenter, NRR
M. Malloy, NRR
G. Mencinsky, NRR

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1125, Volume 23
2. TITLE AND SUBTITLE A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 2001 Annual	3. DATE REPORT PUBLISHED MONTH: August YEAR: 2002 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Compilation 7. PERIOD COVERED *(Inclusive Dates)* Jan. thru Dec. 2001

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address, if contractor, provide name and mailing address.)*

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 54 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2001. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 4, is included by reference only. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
Nuclear Reactors Nuclear Reactor Safety Reactor Operations Safety Engineering Safety Research	Unlimited 14. SECURITY CLASSIFICATION *(This Page)* Unclassified *(This Report)* Unclassified 15. NUMBER OF PAGES 16. PRICE

NRC FORM 335 (2-89)

This form was electronically produced by Elite Federal Forms Inc.