2. Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.
4. U.S. Nuclear Regulatory Commission, NUREG-1560, Volume 1, Part 1, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Summary Report, Draft Report for Comment, October 1996.
5. U.S. Nuclear Regulatory Commission Draft Regulatory Guide, Draft DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated February 28, 1997 (Predecisional).
6. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan Chapter 19, Revision L, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," dated March 3, 1997 (Predecisional).

ATTACHMENT 1

Risk-Based Regulatory Acceptance Criteria
for
Site Specific Application of Safety Goals

T.S. Kress

The purpose of this discussion paper is to explore the concept of using the Safety Goal quantitative health effects (QHO) on early fatalities to derive lower tier risk acceptance criteria for application on a plant-specific basis. A starting point for expressing the early fatality QHO in a form that can be used to derive different tier criteria is the following working definition for the risk of early fatalities for any specific plant in terms of the normal determinations of probabilistic risk assessments (PRAs).

$$\text{Mean number of early fatalities} \equiv \sum_i (CDF)_{i,k}(CCFP)_{i,k}(C_{ef})_{i,k} \qquad (1)$$

where
- k — refers to a specific plant,
- i — refers to the spectrum of accident sequences,
- $(CDF)_{i,k}$ — is the core damage frequency for sequence i of plant k,
- $(CCFP)_{i,k}$ — is the conditional containment failure probability for sequence i and plant k,
- $(C_{ef})_{i,k}$ — is the early fatality consequences at site k given the sequence I which has associated with it a source term St_i, that may be defined in terms of the equivalent release of iodine *to the outside environment*.

The QHO objective for early fatalities is expressed in terms of individual risk. The Safety Goal Policy Statement specifically states that the early fatality QHO is to be determined by calculating the cumulative individual fatalities *within one mile* of the site boundary, C_{ef1}, and dividing that by the population within that same one mile region, P_{k1}. Therefore, Equation 1, for purposes of comparing with the early fatality QHO, should be rewritten as

$$\text{(IR) Individual risk} \equiv \sum_i (CDF)_{i,k}(CCFP)_{i,k}(C_{ef1})_{i,k}/P_{k1} \qquad (2)$$

In order to proceed further, we first note that, in general, the early fatality consequences within one mile of the site boundary can be related to the source term expressed in terms of the equivalent release of iodine by the relationship

$$C_{ef1} \sim (St)^n$$

An appropriate exponent is n = 0.9, which can be obtained from Figure 2 of Attachment 2. Consequently, if a calculation were available that gave the expected early fatalities within one mile of the site boundary, C_{k1r}, for any reference source term, St_r, then Equation 2 would be rewritten as

$$IR \equiv \sum_{i} (CDF)_{i,k}(CCFP)_{i,k} \quad [\frac{C_{klr}}{P_{kl}} (\frac{St_{ik}}{St_r})^n] \tag{3}$$

For our present purposes, Equation 3 can be rewritten as

$$\sum_{i} (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n \equiv \frac{(IR)\ P_{kl}\ (St_r)^n}{C_{klr}} \tag{4}$$

The items on the left of Equation 4 are those that are determined by a full-scope Level 2 PRA with source term (to the environment) capability. The items on the right contain the result of a Level 3 consequence analysis for IR and a site characterization parameter, $(P_{k1})(St_r)^n/C_{k1r}$. This parameter can easily be determined using an appropriate "consequence" code such as MACCS or CRAC and, in fact, may already exist as part of NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

NUREG/CR-2239 basically has a characterization of each US site with respect to C_{ef}. The CRAC code was used to determine site-specific early fatality consequences for

- a standard 1120 Mwe plant (for inventory),
- summary evacuation assumptions,
- actual site population and wind rose,
- best-estimate meteorology, and
- a variety of source terms.

Although the values for P_{k1} and C_{k1r} out to the one mile boundary were not specifically reported in NUREG/CR-2239, the information may still exist in the archival print-outs of the computed output. In case these data are not currently available, Attachment 2 presents a convenient and robust method for estimating the prompt fatalities at each site for any reference source term and provides an appropriate definition of LERF.

Direct Comparison with the QHO Using Level 2 PRAs with Source Term Capability

Equation 4 is an abbreviated form of the early fatality IR that can be used to derive an acceptance criterion for plants with full-scope, Level 2 PRAs with source term capability. For the QHO criterion to be met, the IR must be $\leq 5 \times 10^{-7}$/yr. Therefore the acceptance criterion parameter is clearly

$$\sum_{i} (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n \leq \frac{(5x10^{-7})\ P_{kl}\ (St_r)^n}{C_{klr}} \tag{5}$$

The criterion, Equation 5, can also be cast in terms of a conditional early release probability criterion, CERP, that can be defined as

$$[\sum (CDF)_{i,k}]\ x\ CERP \equiv \sum (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n$$

or

$$CERP_k \equiv \frac{\sum (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n}{\sum_i (CDF)_{i,k}}$$

In terms of acceptance criteria similar to what was recently proposed by the staff with coordinates of CDF and LERF with a basic tenet that CDF should be $\leq 1 \times 10^{-4}$/yr, the appropriate plant specific CERP acceptance criterion would be

$$CERP_{ka} \leq \frac{(5x10^{-7})(St_r)^n(P_{kl})}{(10^{-4})C_{klr}}$$

If a single $CERP_a$ is desired that would bound all current sites, the site with minimum P_{k1}/C_{k1r} would have to be chosen.

Level 2 PRA Without Source Term Capability

For the case of a plant that has Level 2 PRA capability to determine CCFP but not to establish the various associated sequence source terms, it is desired to develop a lower tier criterion in terms of a defined CCFP that would be bounding with respect to ensuring that the early fatality QHO is met on a plant-specific basis.

To do this from Equation 5, it will be necessary to use a single source term that is sure to bound each sequence source term. A possible value that would be bounding (for PWRs only) is 0.5 (release fraction of iodine equivalent). With this value, Equation 5 as a PWR plant-specific bounding criterion becomes

$$\sum_i (CDF)_{i,k}(CCFP)_{i,k} \leq \frac{(5x10^{-7})(P_{kl})(St_r)^n}{(C_{klr}) \quad (.5)^n}$$

The parameter on the left is what I believe the staff used in its decision chart and called a "large early release frequency" (LERF). This would more aptly be called a containment failure frequency.

To express this in terms of a conditional containment failure probability, we could use the usual definition of CCFP.

$$CCFP_k \equiv \frac{\sum (CDF)_{i,k}(CCFP)_{i,k}}{\sum (CDF)_{i,k}}$$

Then, an acceptance value (for PWRs only) for use in a "decision" chart along with CDF = 10^{-4}/yr becomes

$$CCFP_{ka} \leq \frac{(5x10^{-7})}{(10^{-4})} \frac{(P_{kl})}{(C_{klr})} \frac{(St_r)^n}{(.5)^n}$$

A separate acceptance criterion could be developed for BWRs using a bounding St that would have to recognize the effects of the suppression pool and its bypass. An optional way to account for the differences between BWRs and PWRs could be to make different choices for the sequences to go into calculating the CCFP. This is what the staff chose to do - thereby preserving the same acceptance criterion for both reactor types.

For site bounding values, the minimum P_{k1}/C_{k1r} for each reactor type would be chosen. It would be this bounding value that should, in the case of PWRs, be compared with the CCFP = 0.1 that is often mentioned as an appropriate criterion to use along with CDF = 1 x 10^{-4} /yr.

Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA

Rick Sherry- Senior Fellow
Advisory Committee on Reactor Safegua~ds

Introduction

This paper defines a simple relationship between the Safety Goal Quantitative Health Objective (QHO) for Individual Early Fatality Frequency (IEFF) and a Large, Early Release Frequency (LERF) that can be used to estimate the Safety Goal QHO for a specific plant at a specific site. This paper also provides a quantitative definition of the LERF. The relationship utilizes simple site-specific characteristics (wind direction frequencies and population demographics) and results from a Level 2 plant-specific probabilistic risk assessment (PRA) (release category frequencies and source term characteristics).

Estimates using this methodology have been compared with results from detailed calculations performed using the MELCOR Accident Consequence Code System (MACCS) (Ref. 1) for the NUREG-1150 study (Ref. 2), the Grand Gulf Shutdown study (Ref. 3) and the Maine Yankee PRA/IPE. This comparison includes internally initiated sequences, seismic sequences, and shutdown sequences.

Summary of Methodology

The relationship between the IEFF Safety Goal QHO and the LERF is defined as:

$$IEFF = LERF \times EI \qquad \text{Equation 1}$$

$$\text{where: } EI = \text{Exposure Index} = \frac{\sum_{i=1}^{16} P_i \times F_i}{\sum_{i=1}^{16} P_i} \qquad \text{Equation 2}$$

F_i = *the relative frequency wind blows toward sector i*

P_i = *population in sector i within one mile of the plant*

$$and: \quad LERF = \sum_{k=1}^{N_{rc}} RC_k(\text{: early } I > 10\% \text{ and Evacuation Delayed}) \times (1 - F_{no-evac})$$

$$+ \sum_{k=1}^{N_{rc}} RC_k(\text{: total } I > 10\%) \times F_{no-evac} \qquad \text{Equation 3}$$

RC_k = *the frequency of release category k*

N_{rc} = *number of release categories*

$F_{no-evac}$ = *the population fraction not evacuating*

Discussion

The Exposure Index (EI) provides a measure of the average probability that a specific individual within one mile of the plant would be exposed to a lethal radiation dose (given that a release occurs from the plant of sufficient magnitude to produce lethal doses) and assuming that the individual does not evacuate. The exposure index defined above is a slight variation from that initially developed for the Generic Environmental Impact Statement for License Renewal of Nuclear Plants(Ref. 4). Table 1 shows the calculated values for the exposure index for a number of nuclear plant sites using site specific wind direction frequency and population data. Note that all calculated values of the EI fall well within a factor of two of the nominal value of 1/16 which would result if the population within one mile of the plant and the wind direction frequency were uniformly distributed in the sixteen wind direction sectors.

In the above definitions, the release categories which are included in the summations are either 1) those with an early release fraction of iodine greater than 10% of the core inventory and for which evacuation is delayed or 2) release categories where the total iodine release fraction (sum of early and later releases) exceeds 10% and evacuation does not occur. The second summation accounts for the fraction of individuals that do not evacuate and are assumed to be exposed to all time phases of the radionuclide release. This second summation also governs severe seismic sequences where evacuation is assumed to be ineffective.

A ten percent iodine release fraction has been selected as the threshold for source term magnitudes which can result in lethal doses. The selection of 10% was based principally on the work of Kaiser (Ref. 5) which indicated that a threshold for early fatalities occurs at a release fraction of approximately 10% of volatile fission product (I, Cs, Te) release fraction. Figure 1 from Kaiser illustrates this behavior.

An early release is defined as the release which occurs at the time of containment failure (assuming core damage has occurred prior to containment failure). For sequences where containment integrity has been lost prior to core damage, early release begins when core damage

commences. This definition for early release is identical to that provided in NUREG-1150. Typical periods of release duration for early release in the NUREG-1150 study are from several minutes to several hours.

Regardless of the magnitude of the source term, if evacuation commences sufficiently prior to the time when the release of radionuclides begins then the probability of early fatalities is dramatically reduced. In the NUREG-1150 study, three release category timing subgroups were defined for each release category. For subgroup 1, it was assumed that evacuation commenced at least 30 minutes prior to the start of radionuclide release. For subgroup 3, the start of evacuation was delayed until one hour or more after radionuclide release had begun. For subgroup 2, the start of evacuation was assumed to occur within a time window from 30 minutes before, to one hour after, the start of release.

Figure 2 illustrates the impact of these various evacuation assumptions on the Conditional Individual Early Fatality Probability[1] (CIEFP). This figure plots the CIEFP against the iodine release fraction. Individual data points for the three release category subgroups are shown with different symbols. This figure illustrates the effectiveness of early evacuation in reducing the Individual Early Fatality Risk. The diamond shaped symbols represent sequences for which evacuation was delayed until one hour or later after the start of radionuclide release (subgroup 3). For these sequences, the peak CIEFP is on the order of $3 \times (10)^{-2}$. The triangle shaped symbols represent sequences for which evacuation was initiated at least 30 minutes prior to the start of the release of radionuclides (subgroup 1). These sequences have a CIEFP of order $2 \times (10)^{-4}$ or less. These results are dominated by the fraction of the affected population who are assumed not to evacuate. For sequences characterized by evacuation commencing at about the same time as the start of radionuclide release (subgroup 2), the results (shown as circles) generally fall between the results for subgroups 1 and 3.

Equation 1 can be rearranged and used to estimate a plant specific LERF subsidiary objective:

$$LERF = \frac{IEFF}{EI} \qquad \textit{Equation 4}$$

This equation along with the plant specific Exposure Index values shown in Table 1 have been used to calculate a plant specific LERF objective. These results are also shown in Table 1. Note

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1 The CIEFP is conditional on the occurrence of sequences (release categories) with the indicated volatile fission product release magnitude.

that for these calculations, the assumed value for the Safety Goal QHO IEFF is taken as $3 \times (10)^{-7}$ per year.[2]

Example Application Results

The simple methodology has been applied to six cases. These cases are:

Surry Internal Event Initiated Sequences (NUREG-1150)
Surry Seismic Sequences (NUREG-1150)
Peach Bottom Internal Event Initiated Sequences (NUREG-1150)
Peach Bottom Seismic Sequences (NUREG-1150)
Grand Gulf Shutdown Sequences (NUREG/CR-6143)(Ref. 3)
Maine Yankee Internally Initiated Sequences (PRA/IPE submittal)

The results of the example application are summarized in Figure 3. In this figure, the Individual Early Fatality Risk Frequency calculated using the simple methodology is compared with the mean and 95th percentile values from the PRA offsite consequence (MACCS) code calculations. In all cases, the simple methodology produces estimates for the IEFF that are above the PRA mean values. For the two seismic cases, the results fall between the PRA mean and 95th percentile values, and in the remaining cases the results lie above the 95th percentile(but within a factor of three of the 95th percentile value).

The limited comparison summarized in Figure 3 indicates that this methodology may provide a simple and easy to use approach for providing reasonably robust estimates for the IEFF for PRA analyses lacking a detailed Level 3 offsite consequence. This methodology has been applied to a broad spectrum of PRA sequence classes and in all cases the comparison with the PRA results have been favorable.

Discussion of Results

Internally Initiated Sequences (NUREG-1150)

For Peach Bottom internally initiated sequences, all release categories that met the criteria for early iodine release greater than 10% also had evacuation beginning prior to the start of release. This is the principal reason that the MACCS calculated IEFF is low for Peach Bottom. It is only the small fraction of the population that does not evacuate which contributes to the calculated

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2 Based on data from the National Center for Health Statistics the total U.S. societal accident fatality rate was 34.4 and 34.6 deaths per 100,000 individuals in 1993 and 1994, respectively. This translates into an individual early fatality Safety Goal QHO of $3 \times (10)^{-7}$ per reactor year.

IEFF. Consequently, in this study the estimated IEFF for Peach Bottom is controlled by the second summation term in Equation 3 (as for the NUREG-1150 MACCS analyses it was assumed that 0.5% of the population did not evacuate and $F_{no-evac}$ was set to this value).

For the Surry plant, the dominant contributors to a large early fission product release and early fatalities for internally initiated sequences were containment bypass events (12% of CDF). These sequences contribute about 80% of the early fatality risk. For the fast developing interfacing system LOCA bypass sequences, warning times were typically within one hour of the start of radionuclide release. Consequently, evacuation would generally begin after the start of release.

Seismic Sequences (NUREG-1150)

The NUREG-1150 results for seismic sequences for the Peach Bottom and Surry plants using the LLNL seismic hazard curves were compared for this study. The NUREG-1150 seismic sequence consequence analyses were separated into two categories - low Peak Ground Acceleration (PGA)(< 0.6 g) and high PGA (> 0.6 g). For seismic sequences, the offsite emergency protective action assumptions input into the MACCS analyses differed from that for internally initiated events. Particularly important was the assumption that for the high PGA category sequences evacuation was assumed to be ineffective (i.e., does not occur). Consequently, for this study the total iodine release fraction for high PGA seismic sequence classes was used (i.e. the second summation in Equation 3 with $F_{no-evac}$ set to one).

Grand Gulf Shutdown Sequences (NUREG/CR-6143)

For shutdown sequences, the activity level of important volatile fission product species will be reduced by decay from their full power levels and this reduction should be accounted for in analyses. However, for this application this was not done. Even with this limitation the estimated IEFF using the simple methodology was in good agreement with the MACCS results. As for the Peach Bottom internally initiated sequences, warning times for Grand Gulf shutdown sequences were sufficiently early that evacuation commenced prior to the beginning of fission product release, and it is only that small fraction of the population that does not evacuate which contributes to the calculated IEFF. Consequently, the estimated IEFF is controlled by the second summation term in Equation 3.

Maine Yankee Internally Initiated Sequences (PRA/IPE)

The IEFF for Maine Yankee is dominated by sequences where containment failure occurs as a result of a hydrogen combustion event near the time of reactor vessel meltthrough. This type of failure was predicted to occur for about 8% of the core damage frequency. For those sequences contributing to the IEFF, the radionuclide release is predicted to begin about one hour after a general emergency would be expected to be declared. In the Maine Yankee PRA, delay times

ranging from 1 to 2.5 hours were assumed (dependent on the time of year). Hence, for these sequences, radionuclide release generally begins prior to, or coincident with, the start of evacuation.

Enhancements - Use of Iodine Equivalent

Although this paper has shown that reasonably good correlation exists between the IEFF and the LERF calculated using Equation 3 (which uses an iodine release fraction of 10% as threshold for early fatalities), it was also observed that under certain conditions other fission product species groups can make significant contributions to the early fatality risk. It was observed that if the Ru or La group release fractions exceeded about 1 to 5% of core inventory, they began to make a significant contribution to early fatality risk. Furthermore, as discussed above, for sequences initiated during shutdown which may have significant decay periods, the contribution of iodine isotopes to the early fatality risk will be significantly reduced and other radionuclide groups will begin to dominate. For these reasons, a parameter that can be evaluated from the release fraction for all radionuclide groups (and as a function of shutdown time) is desirable. An example of the type of iodine equivalent parameter that is required is presented in NUREG/CR-5164 (Ref. 6) which is based on work reported in Reference 7. In these studies, weighting factors for the contribution to early fatality risk for each radionuclide species group were developed. These weighting factors (which are multiplied by the radionuclide group release fraction) can be used to develop a total iodine equivalent release.

References

1. Chanin, D. I., et al., "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691,Vol. 1, Sandia National Laboratories, February 1990.

2. "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990. [Note, references to NUREG-1150 also imply reference to the NUREG-1150 supporting documents including applicable volumes of NUREG/CR-4551]

3. Brown T. D. et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, Vol. 6, Part 1, March 1995.

4. "Generic Environmental Impact Statement for License Renewal of Nuclear Plants", NUREG-1437, Vol. 1, May 1996.

5. G.D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," ANS Executive Conference on the Ramifications of the Source Term, Charleston, SC, March 12, 1985.

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Nuclear Reactor Accidents'" NUREG/CR-5164, Brookhaven National Laboratory, July 1988.

7. Alpert J., et al, "Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions," NUREG/CR-4467, Sandia National Laboratories, March 1986.

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Table 1
Example of Plant Specific Exposure Index and Derived Subsidiary LERF Objective

Plant	EI	Derived Subsidiary LERF Objective
Grand Gulf	.065	5E-6
Surry	.077	4E-6
Sequoyah	.045	7E-6
Maine Yankee	.040	8E-6
Zion	.081	4E-6
Peach Bottom	.075	4E-6
Lasalle	.083	4E-6
Nominal	.063 (1/16)	5E-6

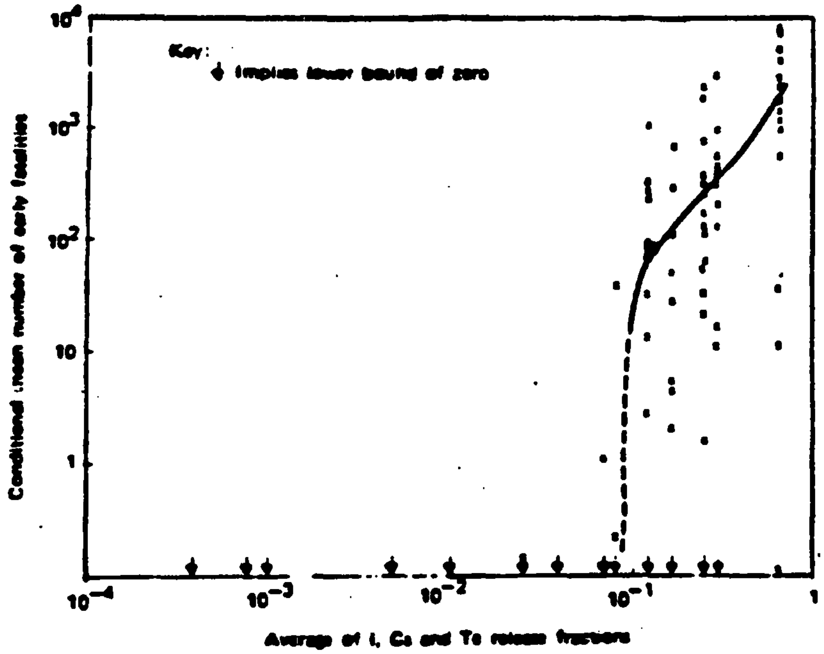
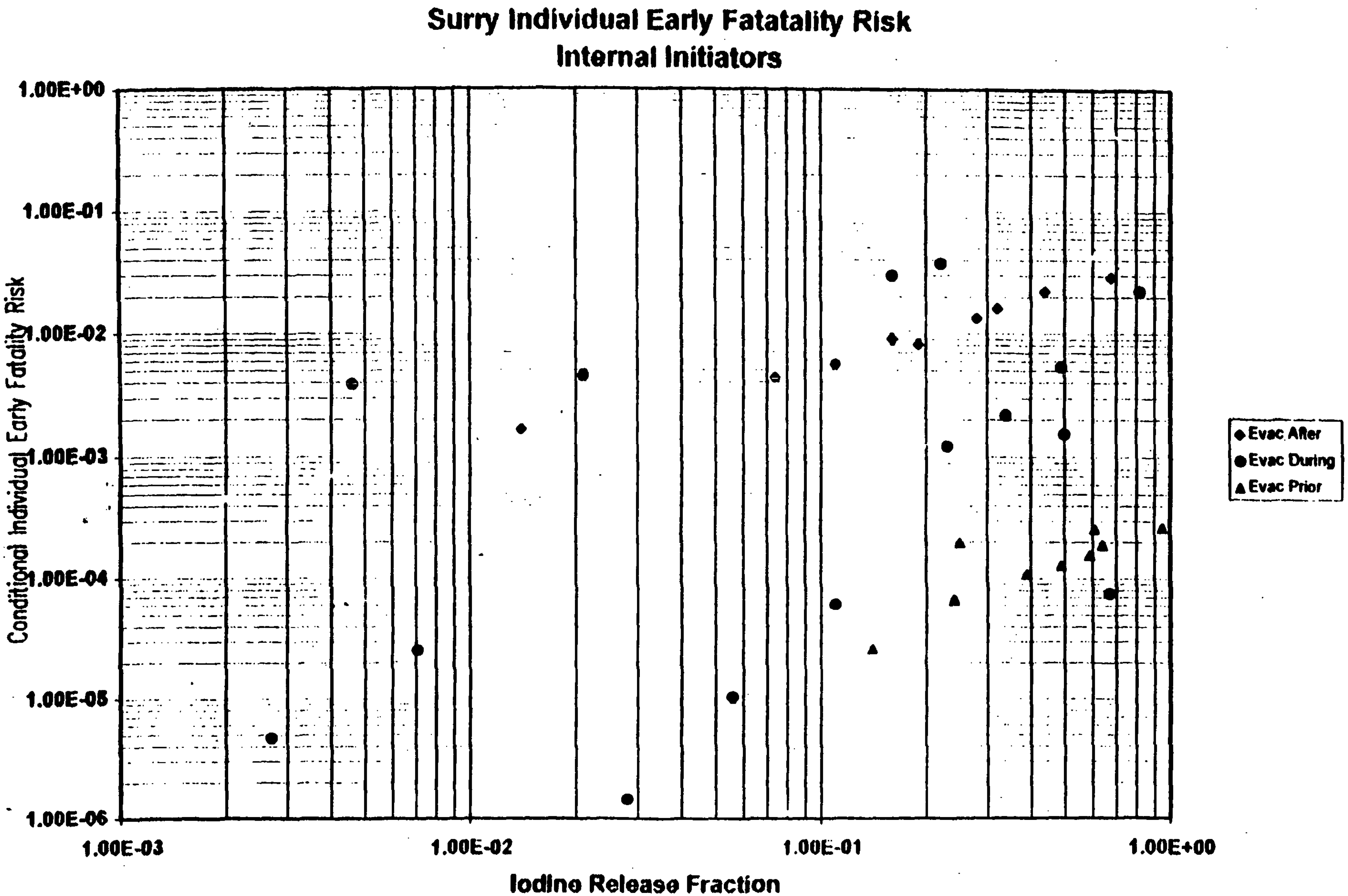


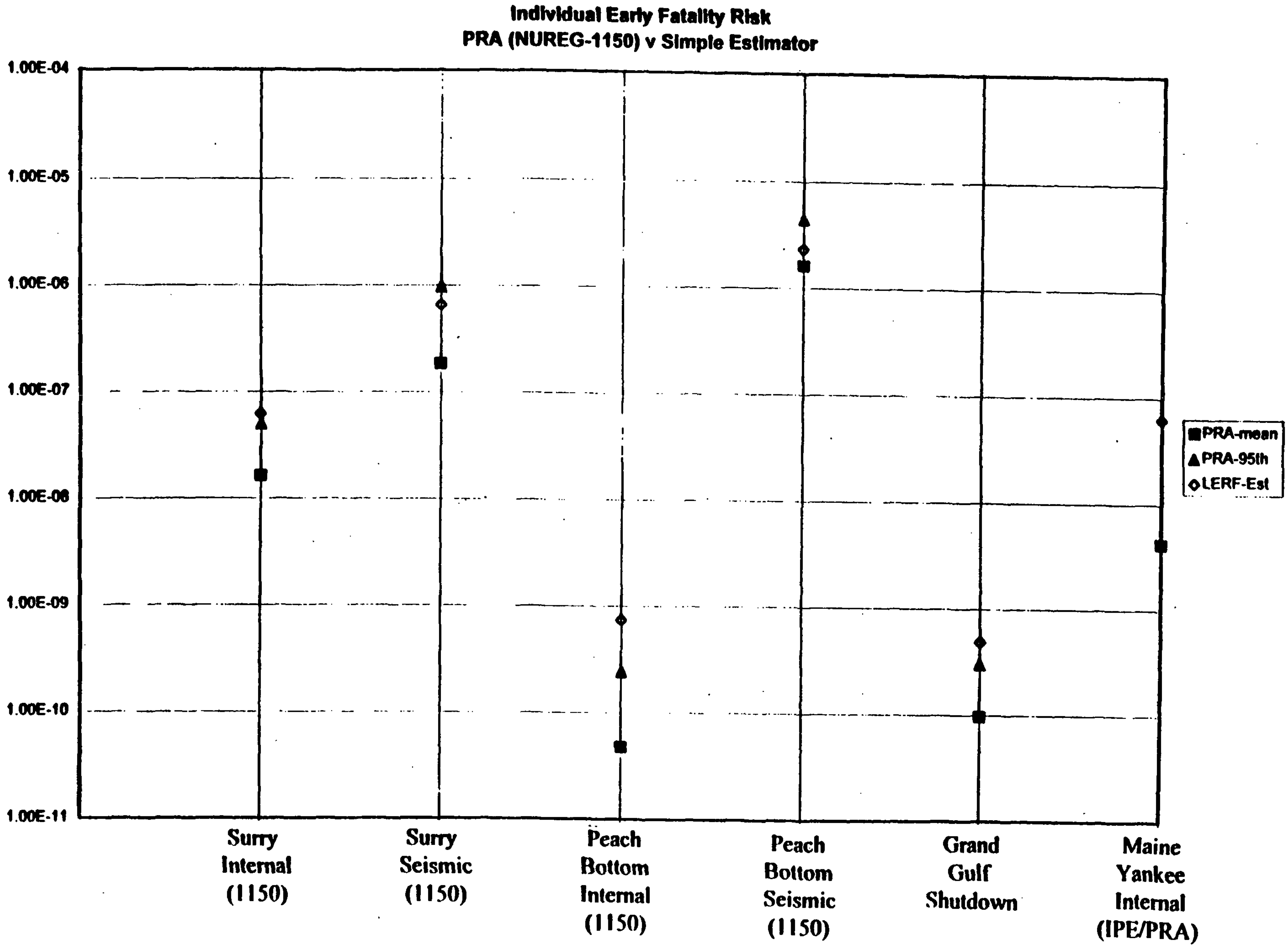
Figure 1. Early Fatalities

Figure 2



Surry Individual Early Fatatality Risk
Internal Initiators
1.00E+00
1.00E-01
1.00E-02
1.00E-03
1.00E-04
1.00E-05
1.00E-06
Conditional Individual Early Fatality Risk
1.00E-03
1.00E-02
1.00E-01
1.00E+00
Iodine Release Fraction
Evac After
Evac During
Evac Prior

Figure 3


Individual Early Fatality Risk
PRA (NUREG-1150) v Simple Estimator
1.00E-04
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
PRA-mean
PRA-95th
LERF-Est
Surry Internal (1150)
Surry Seismic (1150)
Peach Bottom Internal (1150)
Peach Bottom Seismic (1150)
Grand Gulf Shutdown
Maine Yankee Internal (IPE/PRA)

ITEM B.2:

PROPOSED REGULATORY APPROACH ASSOCIATED WITH STEAM GENERATOR INTEGRITY

(DR. SEALE)

ITEM B.2: PROPOSED REGULATORY APPROACH ASSOCIATED WITH STEAM GENERATOR INTEGRITY

On September 12, 1994, based on its review of the draft generic letter concerning the proposed voltage-based repair criteria for Westinghouse steam generator tubes, the Committee provided a report to the Commission. In this report, the Committee agreed with the staff position that rulemaking was the preferred regulatory approach to the problem of steam generator tube degradation, but was skeptical that a new rule could be developed as expediously as specified by the proposed staff schedule.

The Committee reviewed the proposed steam generator rule and an associated regulatory guide and provided a letter to the Executive Director for Operations (EDO) on November 20, 1996. Some of the comments and recommendations in this letter included the following:

- In its present form, the rule is a performance-based regulation almost completely divorced from any direct relation to risk objectives. Such a performance-based rule proliferates the incoherence problems of the present deterministic approach.

- To be risk informed and performance based, the regulatory guide should begin with a clear statement of its objectives, followed by a statement of the performance criteria and the guidelines for meeting the criteria.

- The Committee agrees that the staff should approve the performance criteria that are proposed by licensees to implement the steam generator rule. Industry, however, should be provided more flexibility to propose alternative performance criteria supported by an appropriate risk analysis.

- The staff position is that the regulatory guide provides sufficient guidance for developing an acceptable methodology and that formal review of industry-developed repair criteria and procedures will not be required.

- The staff should prepare a point-by-point response to the outstanding issues in the differing professional opinion, and resolve generic safety issue GSI-163, "Multiple Steam Generator Tube Leakage," before implementing the steam generator integrity rule.

The EDO responded to the above ACRS letter on January 2, 1997. In this letter, the EDO suggested holding additional meetings to discuss the Committee comments and recommendations. Following a meeting between the Committee and the staff on January 9, 1997, the Executive Director for the ACRS issued a memorandum to the staff on January 31, 1997, as suggested by the Committee, stating that the ACRS request that the staff, at the next ACRS Subcommittee meeting, respond to the comments and recommendations included in the November 20, 1996 ACRS letter and to several concerns raised by individual Committee members. Some of the individual member's concerns are as follows:

- The performance criteria, which are intended to ensure that the risk due to thermally induced tube failures during severe accidents is acceptable, appear to introduce new requirements. The basis for the introduction of these requirements is unclear. Very little explanation has been provided that clearly identifies how the performance criteria and program requirements are related to risk analyses.

- The staff should clarify whether performance criteria are derived from top level risk requirements or from defense-in-depth considerations. The staff should identify the number of and bases for the defense-in-depth criteria.

- The regulatory guide does not describe, in all cases, a standard for complying with the performance criteria or program requirements.

- The regulatory guide does not, in all cases, identify how the licensees can demonstrate that reasonable assurance has been achieved.

The ACRS Subcommittees on Materials & Metallurgy and on Severe Accidents held a joint meeting on March 4-5, 1997, with the NRC staff to discuss the draft risk assessment of severe accident induced steam generator tube ruptures, and the associated draft regulatory analysis. The staff stated that based on the results of these two efforts, it was reassessing its regulatory approach for addressing steam generator tube integrity issues. Specifically, the staff is reexamining alternative regulatory approaches for addressing steam generator tube integrity. These include utilizing the current regulations and the PRA Implementation plan as a framework for regulating steam generator tube integrity issues rather than a new rule.

The staff is drafting a memorandum to the Commission that will discuss the general conclusions from the draft risk assessment, the major conclusions from the draft regulatory analysis, the implications for steam generator rulemaking, and an alternative regulatory approach to accomplish the original objectives of the steam generator rule. The Committee plans to review and comment on the staff approach after receiving the staff memorandum.

Attachments:

- Letter dated November 20, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Steam Generator Integrity (pp. 41-45)

- Letter dated January 2, 1997, from James M. Taylor, Executive Director for Operations, NRC, to Thomas S. Kress, Chairman, ACRS, Subject: Staff Response to ACRS Comments on Proposed Steam Generator Rule (pp. 46-47)

- Memorandum dated January 31, 1997, from John T. Larkins, Executive Director, ACRS, to Ashok C. Thadani, Office of Nuclear Reactor Regulation, Subject: Steam Generator Integrity Rulemaking (pp. 48-50)

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

November 20, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE ON STEAM GENERATOR INTEGRITY

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we reviewed the technical bases for the proposed steam generator integrity rule and an associated regulatory guide. During the 432nd meeting of the ACRS, June 12-14, 1996, and meetings of the Joint Subcommittees on Materials & Metallurgy and on Severe Accidents, June 3-4 and November 5-6, 1996, we heard presentations on subjects related to this matter. During these reviews, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute, and the Electric Power Research Institute, as well as the author of a differing professional opinion. We also had the benefit of the documents referenced.

The proposed steam generator integrity rule is intended to provide a risk-informed and performance-based regulation to replace an existing prescriptive regulation. In its present form, the rule is a performance-based regulation almost completely divorced from any direct relation to risk objectives. Such a performance-based rule proliferates the incoherence problems of the present deterministic approach. The proposed rule preserves a tenuous connection between "design-basis space" and "risk space" without clearly articulating the risk objectives.

Some of the characteristics exhibited in the development process of the rule and regulatory guide include the following:

- difficulty in reaching agreement on the performance criteria,
- incomplete and sometimes perfunctory analyses required to provide an assessment of relative risk,
- reliance on core-damage frequency alone as an indicator of risk, and

- recourse to defense-in-depth without specific criteria for its use.

We believe that more direct consideration of risk could have avoided some of these difficulties.

A controversial element of the proposed rule and regulatory guide is the introduction of severe accident issues into an area that has been exclusively resolved by using a design-basis analysis. This extension of the scope of accident analysis is necessary to make risk-informed regulatory decisions and is part of the cost of moving toward risk-informed regulation. Since licensees have done risk-informed analyses for the Individual Plant Examination (IPE) process, we believe that the analysis for addressing severe accident events should not be overly burdensome to them.

Steam generator tube ruptures are small contributors to the total core-damage frequency, but may be risk significant due to containment bypass effects. In previous analyses, the staff performed limited assessments of primary side fission product attenuation and neglected secondary side attenuation. The regulatory guide now proposes that the licensees deal with the risk of a thermally induced tube failure either by demonstrating that the frequency of the initiating events is sufficiently low (10^{-6}/reactor year) or by demonstrating that the conditional probability of tube failure, given that an initiating event has occurred, is low (on the order of 0.1). We believe that licensees should also be given the option to demonstrate that, even if thermally induced tube ruptures occur, the associated risk is low when a more realistic treatment of fission product attenuation is made.

We are concerned that the proposed regulatory guide, as presented, could send the wrong message to licensees that risk-informed and performance-based requirements are add-ons to the traditional design-basis accident approach and can only result in an additional burden. We believe that to be risk informed and performance based, the regulatory guide should begin with a clear statement of its objectives, followed by a statement of the performance criteria and the guidelines for meeting the criteria. We note that the staff has stated that the proposed performance criteria have been derived from risk analyses, but we have not seen these analyses. Rewriting the regulatory guide is not a trivial task, but could result in a regulatory framework that could be used as a model for future risk-informed and performance-based rulemaking efforts.

In other applications of performance-based regulation such as the Maintenance Rule, the licensees have been permitted to determine appropriate performance criteria and have been given more flexibility in developing the methodology used to determine whether the criteria have been met. For the steam generator rule, the

staff has concluded that it should approve the performance criteria that are proposed by licensees to implement the steam generator rule. We agree with the decision of the staff that it should approve the criteria. Industry, however, should be provided more flexibility to propose alternative performance criteria supported by an appropriate risk analysis. We would like to review all of the supporting documentation before commenting on the specific criteria that have been proposed in the regulatory guide.

The demonstration that the criteria have actually been satisfied requires a complex process of nondestructive examination and evaluation of structural integrity and leakage during operation and design-basis accidents. The methodology required for these evaluations is not well established. Thus, the staff has felt constrained to provide a great deal of detail in the proposed regulatory guide to describe the characteristics of an acceptable methodology. Although we are not yet prepared to endorse the regulatory guide, we believe that the present immaturity of the methodology and the importance of the results justify such an approach.

The staff position is that the regulatory guide provides sufficient guidance for developing an acceptable methodology and that formal review of industry-developed repair criteria and procedures will not be required. We would like to review the results of a "trade study" of the preapproval approach vs. the post-implementation inspection approach to methodology acceptance.

Industry has questioned whether safety factors proposed in the steam generator rule are more conservative than those required by the ASME code. We encourage the staff to consider the industry's arguments.

Industry accepts the performance criterion proposed by the staff for primary-to-secondary leakage. Industry stated that this leakage criterion ought not be *ipso facto* a trigger for inspection or enforcement of regulations concerning the steam generator rule. This is a valid concern. Excessive leakage does not necessarily indicate a failure of the steam generator program. Adequate opportunities for staff action are available if failures of the program are discovered following a plant shutdown due to excessive primary-to-secondary leakage.

We are looking forward to reviewing the staff NUREG report concerning the staff's treatment of thermally induced tube failure. We are especially interested in the treatment of elevated temperatures resulting from flow through leaking tubes, and coupling between aerosol deposition and thermal hydraulics.

A differing professional opinion (DPO) was filed on July 11, 1994. We have reviewed the contentions in that DPO and summarized them in

the attachment. We also note that Generic Safety Issue (GSI)-163, "Multiple Steam Generator Tube Leakage," identified in 1992 has yet

to be prioritized and resolved. Both the DPO and the GSI are directly related to the proposed rulemaking. We urge the staff to prepare a point-by-point response to the issues in the DPO and to prioritize and resolve GSI-163 before implementing the steam generator integrity rule.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Attachment:
Summary of Differing Professional Opinion Issues - Presented to the ACRS on November 7, 1996

<u>References</u>:

1. Memorandum dated October 25, 1996, from Brian Sheron, Office of Nuclear Reactor Regulation, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of the Proposed Steam Generator Rule [forwarding the proposed steam generator rule and draft steam generator regulatory guide]
2. Memorandum dated May 1, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, Subject: Resolution of Differing Professional Opinion Regarding Voltage-Based Repair Criteria for Steam Generator Tubes, dated July 13, 1994
3. Memorandum dated July 15, 1994, from James M. Taylor, Executive Director for Operations, NRC, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review Of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes [forwarding Differing Professional Opinion]
4. Report dated September 12, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
5. Memorandum dated September 30, 1994, from Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Comments On ACRS Review Of Generic Letter "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes"

SUMMARY OF DIFFERING PROFESSIONAL OPINION ISSUES PRESENTED TO THE ACRS ON NOVEMBER 7, 1996

The DPO author estimates core-damage frequency with containment bypass to be 10^{-4} - 3.4×10^{-4} events/year. He stated that the uncertainties associated with characterizing steam generator tube defects and severe accident phenomena are not sufficiently understood to properly model tube rupture events. Tubes may fail before the surge line due to:

- crack networking and characterization of flaws not being adequately determined by nondestructive examinations,
- increased heat transfer caused by flow through tube cracks,
- cracks in tubes opening due to increased pressure,
- cracks in tubes unplugging at elevated pressure, and
- jets from tube cracks eroding adjacent tubes.

The DPO author stated that the staff should document the assumptions and models used to study hidden uncertainties.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 2, 1997

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: STAFF RESPONSE TO ACRS COMMENTS ON PROPOSED STEAM GENERATOR RULE

Dear Dr. Kress:

I am responding to your letter of November 20, 1996, in which you forwarded the comments of the Advisory Committee on Reactor Safeguards (ACRS) on the technical bases for the proposed steam generator rule and an associated draft regulatory guide. The staff appreciated the opportunity to brief the ACRS on this matter in November 1996, and the timely response from the committee.

The staff has concluded that additional meetings on the proposed rulemaking are appropriate. In this regard, NRR and ACRS staff are working to schedule several additional meetings. It is our intent that the first of these meetings, to be scheduled in the near future, would provide an opportunity to discuss some broad issues related to implementation of risk-informed regulation. These issues are related not only to the proposed steam generator rule, but also to other risk-informed initiatives that the staff has recently presented to the ACRS. Following the next meeting, we will schedule additional subcommittee and full committee meetings to further discuss the specifics of the proposed steam generator rulemaking.

I would also like to note that it is the staff's intent that the differing professional opinion (DPO) concerns that were presented to the ACRS and summarized in your letter, will be addressed in the rulemaking package.

CONTACT: Timothy A. Reed, NRR
415-1462

I appreciate your cooperation with the staff in resolving these issues so that the agency can proceed with the proposed steam generator rulemaking.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY

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

January 31, 1997

MEMORANDUM TO: Ashok C. Thadani, Acting Deputy Director
Office of Nuclear ~~Reactor~~ Regulation

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

SUBJECT: STEAM GENERATOR INTEGRITY RULEMAKING

During the joint meeting of the ACRS Subcommittees on Materials & Metallurgy and Severe Accidents on January 9, 1997, the staff discussed the broad issues related to risk-informed, performance-based regulation, as well as the status of developing the steam generator integrity rule, the associated regulatory guide, and the technical analyses. This meeting was also intended to discuss the staff response to ACRS comments and recommendations contained in its November 20, 1996 letter to the Executive Director for Operations concerning the proposed steam generator integrity rule and the associated regulatory guide.

In fact, the January 9 meeting was primarily devoted to a discussion of the staff's approach to completing the Standard Review Plan sections and supporting documentation associated with risk-informed, performance-based regulation. In addition, a discussion was held on the approach for developing the steam generator integrity rule, the related regulatory guide, and other documents. Although these discussions were somewhat helpful to the members, this meeting did not completely resolve the comments and recommendations included in the November 20, 1996 ACRS letter. During the Subcommittee meeting, the staff expressed an intent to incorporate some of the comments and recommendations into the proposed documents; however, the extent to which the ACRS concerns would be reflected in these documents was not clear.

The meeting did allow the members to sharpen the focus of some of their concerns. Many of the concerns of the members are about how the specific performance criteria and guidance described in the regulatory guide relate to risk. Much of the difficulty that the members have had in reviewing the steam generator integrity rule and regulatory guide is due to the unavailability of written documents (e.g., NUREG and regulatory analyses) that describe the bases for the performance criteria. The information provided in general terms by the staff during Subcommittee meetings was

insufficient for the members to clearly understand the bases for such criteria. The Committee requests that the staff, at the next Subcommittee meeting, respond to the ACRS comments and recommendations included in the November 20, 1996 letter and to the following concerns raised by individual Subcommittee members:

- The performance criteria, which are intended to ensure that the risk due to thermally induced tube failures during severe accidents is acceptable, appear to introduce new requirements. The basis for the introduction of these requirements is unclear. Very little explanation has been provided that clearly identifies how the performance criteria and program requirements are related to risk analyses.

- The staff should explain how the proposed rule will transition from the existing deterministic regulatory process, which is based on design-basis accident information, to a risk-informed, performance-based process, which is based on risk information.

- The regulatory guide does not clearly state the objectives, functional requirements, performance requirements, verifications, acceptable solutions, and alternative solutions associated with the performance criteria.

- The staff should clarify whether performance criteria are derived from top level risk requirements or from defense-in-depth considerations. The staff should identify the number of and bases for the defense-in-depth criteria.

- The regulatory guide does not describe, in all cases, a standard for complying with the performance criteria or program requirements.

- The regulatory guide does not, in all cases, identify how the licensees can demonstrate that reasonable assurance has been achieved.

- The staff should explain the basis for the 0.05 tube failure per year criteria -- in particular, how the value for steam generator tube ruptures is an appropriate allocation of the total conditional containment failure probability; how the criteria for tube plugging are derived from inspection findings, leak rates, or voltage criteria; or how plugging criteria are derived from a probability of tube failure.

- The regulatory guide does not explain how licensees should demonstrate that the spontaneous probability of steam generator tube failures will be below the assumed failure criteria.

- The staff should explain the basis for the allocation of 20 percent of the tube failure probability criteria to each degradation mechanism.

The Committee looks forward to meeting with you and your staff in the near future. If you have any questions or need additional information, please contact Noel Dudley of my staff at 415-6888.

cc: ACRS Members
J. Hoyle, SECY
D. Morrison, RES
J. Mitchell, OEDO
F. Miraglia, NRR
B. Sheron, NRP
J. Cortez, RES

ITEM B.3:

LOW-POWER AND SHUTDOWN OPERATIONS RISK

(DR. POWERS)

ITEM B.3: LOW-POWER AND SHUTDOWN OPERATIONS RISK

The Committee provided a report to the Commission dated April 18, 1997, regarding low-power and shutdown operations risk. The Committee has become concerned about the need for developing an understanding of the risk posed by the low-power and shutdown operations at nuclear power plants. This need arises because of repeated events during these modes of power plant operations, changes being made in plant operations in response to economic forces, and because of the ongoing initiative to develop risk-informed, performance-based regulation at the NRC. The staff will have to gain an understanding of risk during low-power and shutdown operations commensurate with its understanding of risk during power operations.

During the December 1996 ACRS meeting, the Committee heard presentations by and held discussions with representatives of the Office for Analysis and Evaluation of Operational Data (AEOD) concerning the Human Performance Event Database. Information from this database indicated that more than 50 percent of the events investigated by Augmented Inspection Teams have occurred when plants were in low-power or shutdown modes. The evaluations of potential severe accidents during low-power and shutdown conditions at Surry and Grand Gulf show that low-power and shutdown risk is a significant factor compared to the risk calculated for the same plants during power operations. Based on previous evaluations of low-power and shutdown operations risk, and the above facts, the Committee provided the April 18, 1997 report to the Commission. This report contains several comments and recommendations, including the following:

- It is essential for the success of the Commission's efforts to adopt risk-informed, performance-based regulation that a more complete understanding of the full spectrum of risk be established on a defensible technical basis.

- At present, there is no defensible regulatory basis to determine the extent to which results of risk analyses for power operations ought to be augmented to account for risk of low-power and shutdown operations. A more complete understanding of risks during all phases of nuclear plant operations is essential to ensure that regulations address real, significant risks and do not impose ad hoc measures to correct discovered deficiencies in the hope that these measures will also address risk-significant issues.

- Significant efforts may be needed to establish new risk assessment methods and to understand phenomena associated with core damage events and the dispersal of radioactivity during low-power and shutdown operations.

- The NRC staff needs to establish a high-quality benchmark on risk during low-power and shutdown operations comparable to that which it has derived for risk during power operations.

- A well-planned, deliberate effort with realistic time schedules and extensive peer review should be undertaken first to develop methods and technologies that may be needed and then to benchmark risk during low-power and shutdown operations.

The Committee also provided a letter to the Executive Director for Operations on June 4, 1996, which was based on the Committee's review of the proposed rule on shutdown operations and of the shutdown risk studies performed for the Surry and Grand Gulf plants. In this letter the Committee made several comments, including the following:

- The concern for risk associated with shutdown operations has arisen from incidents that have occurred. The quantitative understanding of risk posed by plants in low-power or shutdown modes of operation is limited.

- The available evidence suggests that shutdown operations can make important contributions to the overall risk to the public posed by nuclear power plants. There are no complete, reliable assessments of risk during shutdown operations even for a few representative plants. Certainly, there is nothing commensurate with the NUREG-1150 study of risk during full-power operation.

- The staff effort toward an interim solution by promulgating the proposed shutdown operations rule is based on engineering judgment and will probably lessen risk. A risk-informed understanding will require a quantitative evaluation of risk during low-power and shutdown operations. Therefore, priority attention should be given to performing Level 3 PRAs for shutdown operations for the NUREG-1150 plants with consideration of spent fuel pool risk and uncertainty assessments.

The Committee plans to review the shutdown operations rule, which will also address spent fuel pool operations. The Committee also plans to review the extent to which low-power and shutdown operations risk will be incorporated into the development of risk-informed and performance-based regulations and regulatory guidance.

Attachments:

- Report dated April 18, 1997, from Robert L. Seale, ACRS Chairman, to Shirley Ann Jackson, NRC Chairman, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations (pp. 54-59)

- Letter dated June 4, 1996, from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Shutdown Operations (pp. 60-61)

- Letter dated June 28, 1996, from James M. Taylor, Executive Director for Operations, NRC, to T. S. Kress, ACRS Chairman, Subject: Comments on the Proposed Rule on Shutdown Operations (p. 62)

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

April 18, 1997

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: ESTABLISHING A BENCHMARK ON RISK DURING LOW-POWER AND SHUTDOWN OPERATIONS

This report is to draw attention to the critical need for developing an understanding of risk posed by low-power and shutdown operations at nuclear power plants. This need is apparent as a result of: (1) repeated events during these modes of power plant operations, (2) changes being made in plant operations in response to economic forces, and (3) the ongoing NRC initiative to develop risk-informed, performance-based regulation. We believe it is essential that the NRC staff undertake a quantitative examination of risk during low-power and shutdown operations at representative nuclear power plants. That is, the NRC staff needs to establish a high-quality benchmark on risk during low-power and shutdown operations comparable to that which it has derived for risk during power operations from the NUREG-1150 study [Ref. 1] and other sources. The benchmark for risk during low-power and shutdown operations should address the following:

- a representative range of plant types,
- all phases of low-power and shutdown operations,
- accidents initiated by internal fires and other external events,
- human performance, the unusual source term, radionuclide dispersal, and on-site populations that will affect the predictions of accident consequences, and
- uncertainties to a depth similar to that done for the risk benchmark for power operations.

A substantial effort will be required to develop the technical capabilities to conduct this benchmark risk analysis. Results of the benchmark risk analysis may suggest the need for refinements to the Commission's Safety Goals. In particular, the Commission may find from the results that it wants to specify limits on the tolerable durations of plant configurations that pose very high risks.

Our recommendation for a detailed benchmark analysis is based on the results of scoping risk studies done by the staff contractors [Refs. 2,3], the continuing string of worrisome events at plants during low-power and shutdown operations, and assessments of the risk significance of plant events by the Office for Analysis and Evaluation of Operational Data. The staff's contractors have done limited analyses of risk during one phase of shutdown operations at a pressurized water reactor with a subatmospheric containment [Ref. 2] and one phase of shutdown operations at a Mark III boiling water reactor [Ref. 3]. Results of these studies show that even when the risk for a short period of shutdown operations is normalized over a full calendar year, the risk is a significant fraction of the risk calculated for the same plant during power operations:

Boiling Water Reactor

	Power Operations	Shutdown Operations*
Mean Core Damage Frequency	4.1×10^{-6}	2.1×10^{-6}
Mean Early Fatality Risk	8.2×10^{-9}	1.4×10^{-8}
Mean Latent Cancer Fatality Risk	9.5×10^{-4}	3.8×10^{-3}

* Plant Operating Mode 5 (cold shutdown) only.

Pressurized Water Reactor

	Power Operations	Shutdown Operations*
Mean Core Damage Frequency	4.1×10^{-5}	4.2×10^{-6}
Mean Early Fatality Risk	2.0×10^{-6}	4.9×10^{-8}
Mean Latent Cancer Fatality Risk	5.2×10^{-3}	1.6×10^{-2}

* Mid-loop operation only.

These partial results, however, may not adequately reflect current operating practices. The industry has instituted new guidelines [Ref. 4] for low-power and shutdown operations that are intended to reduce risk. Several licensees are using software such as the Electric Power Research Institute's ORAM (Outage Risk Assessment

and Management) to plan activities during low-power and shutdown operations. These software tools are based on risk insights derived from simplified probabilistic risk assessments. If the NRC staff is to provide effective safety oversight of low-power and shutdown operations, the staff will have to understand the technical bases of the software tools and the approximations in risk assessments that have been used to develop these tools. The availability of benchmark risk assessments for low-power and shutdown operations for representative plants akin to the benchmark risk assessments for power operations appears to be essential for the development of this understanding.

Despite the new guidelines and software tools developed by the nuclear power industry for low-power and shutdown operations, events that reveal safety vulnerabilities of the plants continue to occur. Among the more recent of these events are:

- The Wolf Creek plant was in a "hot shutdown" condition when activities involving the residual heat removal system created a flow path that allowed approximately 9,200 gallons of reactor coolant to transfer to the refueling water storage tank. Had this draining not been promptly terminated, the operability of the emergency core cooling system would have been compromised. The Accident Sequence Precursor Analysis indicated that this event had a high conditional core damage probability [Ref. 5]. The scoping studies of shutdown risk, however, suggested that a "hot shutdown" condition was of such a low risk significance that it did not merit quantification.

- Loss of core cooling was threatened by the formation of a nitrogen bubble in the reactor coolant system at the Haddam Neck plant as a result of an improper valve lineup. Injection of high-pressure nitrogen into the reactor vessel continued for over three days while the plant was in a "cold shutdown" condition. The water level in the reactor vessel was believed to have been displaced three feet below the vessel flange [Ref. 6].

- At the Cooper plant, about 10,000 gallons of water was inadvertently lost from the refueling cavity because a submerged valve was opened to the main steam line drains. It took over an hour for operators to identify the source of the loss of coolant inventory [Ref. 7].

Events during low-power and shutdown operations are consuming significant staff resources. At our meeting on December 5, 1996, we were told that more than 50 percent of recent events requiring Augmented Inspection Teams have occurred when plants were in low-power or shutdown conditions. Human errors during these conditions appear to be especially probable. A number of incidents that have

occurred during low-power and shutdown conditions are reviewed in the report NUREG/CR-6093 [Ref. 8]. This report concluded that factors influencing operator actions are different from those typically regarded as important during full-power operations and states that: "Unlike full-power operations, large numbers of multiple concurrent tasks are possible during LP&S (low-power and shutdown) conditions. This has implications for both the PRA (probabilistic risk assessment) modeling process and the HRA (human reliability assessment) quantification process."

We are concerned that this situation will be exacerbated as the industry moves to longer cycle times with less frequent opportunities to exercise its low-power and shutdown operating procedures. The situation may also be exacerbated by industry efforts to shorten the duration of low-power and shutdown operations by increasing the intensity of activities during these periods. The industry will want to relieve burdens during outages by doing some maintenance while the plant is operating. For the staff to approve a trade-off between maintenance "on-line" and maintenance during outages, it will have to consider risk. To do this, the staff will have to gain an understanding of risk during low-power and shutdown operations commensurate with its understanding of risk during power operations.

The staff is now embarked on an effort to develop risk-acceptance criteria for providing regulatory relief to licensees. Staff judgments on these matters are based on a firm foundation concerning event probabilities during power operations derived both from the Individual Plant Examinations done by licensees and from its own benchmarking risk studies reported in NUREG-1150. There is no comparable basis for making judgments concerning the accident probabilities and risk during low-power and shutdown operations. At present, there is no defensible regulatory basis to determine the extent to which results obtained for power operations ought to be augmented to account for risk of low-power and shutdown operations. A more complete understanding of risks during all phases of nuclear plant operations is essential to ensure that regulations address real, significant risks and do not impose ad hoc measures to correct discovered deficiencies in the hope that these measures will also address risk-significant issues.

We believe it is essential for the success of the Commission's effort to adopt risk-informed, performance-based regulation that a more complete understanding of the full spectrum of risk be established on a defensible technical basis. This more complete understanding is needed now as pivotal decisions are being made on the implementation of risk-informed, performance-based regulation. We do not believe that existing scoping analyses or further scoping efforts will establish adequate benchmarks concerning risk during low-power and shutdown operations. This is especially so in light

of evidence that time-dependent human performance is important. Significant efforts may be needed to establish new risk assessment methods and to understand phenomena associated with core damage events and the dispersal of radioactivity during these phases of plant operations. We are confident that areas of substantial uncertainty will arise in the assessment of risk dur'ng low-power and shutdown operations. Defensible quantification of these uncertainties will require the same type of effort that was needed to quantify uncertainties in risk during power operations.

It will take time to develop a usefully complete understanding of risk during low-power and shutdown operations. We recommend that a well-planned, deliberate effort with realistic time schedules and extensive peer review be undertaken first to develop methods and technologies that may be needed and then to benchmark risk during low-power and shutdown operations.

Sincerely yours,

R. L. Seale

R. L. Seale
Chairman

<u>References</u>:

1. U.S. Nuclear Regulatory Commission, NUREG-1150, Vol. 1, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-6144, BNL-NUREG-52399, Vol. 1, Brookhaven National Laboratory, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Summary of Results, October 1995.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6143, SAND93-2440, Vol.1, Sandia National Laboratories, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," Summary of Results, July 1995.
4. NUMARC 91-06, Nuclear Management and Resources Council, Inc., "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.
5. U.S. Nuclear Regulatory Commission, Information Notice 95-03, Supplement 1, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," dated March 25, 1996.
6. U.S. Nuclear Regulatory Commission, Information Notice 94-36, Supplement 1, "Undetected Accumulation of Gas in Reactor Coolant System," November 1996.

7. U.S. Nuclear Regulatory Commission, AEOD/S96-02, "Assessment of Spent Fuel Pool Cooling," September 1996.

8. U.S. Nuclear Regulatory Commission, NUREG/CR-6093, "An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues," June 1994.

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

June 4, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE ON SHUTDOWN OPERATIONS

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, we held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), concerning the subject proposed rule and the probabilistic risk assessment (PRA) studies that were performed for the Surry and the Grand Gulf Nuclear Power plants. Our Subcommittee on Plant Operations met with the staff, NEI, and a utility representative on May 21, 1996, to discuss these matters. We also had the benefit of the documents referenced. We previously commented on the staff effort to resolve the shutdown operations issue in our letters dated August 13, 1991, April 9, 1992, September 15, 1992, and May 13, 1994.

According to the staff, the proposed rule will contain performance-based elements. Since the supporting regulatory analysis and regulatory guide are still being developed, we discussed only the proposed rule during our meeting. The staff has held several public meetings with NEI to obtain industry input on the formulation of this rule.

We made a number of comments on the risk basis for the rule. The staff agreed to consider our comments as it finalizes the draft rule, which it plans to publish for public comment in September 1996. We plan to provide comments on the proposed final rule after the staff has reconciled the public comments.

The concern for risk associated with shutdown operations has arisen from incidents that have occurred. Our quantitative understanding of the risk posed by plants in low-power or shutdown modes of operation is limited. Risk assessments for shutdown operations were performed for Surry (a three-loop PWR with loop isolation valves and a sub-atmospheric pressure containment) and Grand Gulf (a BWR-6 with a Mark III containment). Neither of these plants is a particularly good surrogate for the entire population of PWRs and BWRs.

The studies of shutdown risk consisted of two phases. The first phase was a deliberately conservative scoping analysis. The second phase focused on a single, high-risk plant operational state among the many that exist during shutdown operation. Such an approach could lead to an incorrect assessment of risk (a historical analogue is the selection of the large-break, loss-of-coolant accident as a bounding event) or to the adoption of operating practices that might increase risk.

The available evidence does suggest that shutdown operations can make important contributions to the overall risk to the public posed by nuclear power plants. On the eve of our entry into an era of risk-informed rulemaking, there are no complete, reliable assessments of risk during shutdown operations even for a few representative plants. Certainly, there is nothing commensurate with the NUREG-1150 study of risk during full-power operation.

The staff effort toward an interim solution by promulgating this proposed rule is based on engineering judgment and will probably lessen risk. A risk-informed understanding will require a quantitative evaluation of risk during low-power and shutdown operations. We therefore recommend that priority attention be given to performing Level 3 PRAs for shutdown operations at the NUREG-1150 plants with consideration of spent fuel pool risk and uncertainty assessments.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated April 5, 1996, from Robert C. Jones, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, Subject: Development of §50.67, "Shutdown Operation of Nuclear Power Plants"
2. U. S. Nuclear Regulatory Commission, Prepared by Brookhaven National Laboratory, NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Summary of Results, October 1995
3. U. S. Nuclear Regulatory Commission, Prepared by Sandia National Laboratories, NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," Summary of Results, July 1995
4. Nuclear Management and Resources Council, Inc., NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 28, 1996

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: YOUR COMMENTS ON THE PROPOSED RULE ON SHUTDOWN OPERATIONS (TAC NO. M77701)

Dear Dr. Kress:

Thank you for the comments in your letter of June 4, 1996, on the proposed rule §50.67, "Shutdown Operation of Nuclear Power Plants." We agree that available evidence does suggest that shutdown operations can be important contributors to the overall risk to the public. We also recognize that the risk assessments for the Surry and the Grand Gulf plants cannot be used as surrogates for the entire population of pressurized water reactors and boiling water reactors. Nonetheless, we are using those insights which are generic to supplement our deterministic assessments in performing the regulatory analysis.

You noted that the proposed rule is based significantly on engineering judgment. While this is true, this judgment draws upon many years of experience with shutdown operations, a good understanding of events that have occurred and the causes of those events, and accounts for those shutdown activities which the staff's shutdown study, documented in NUREG-1449, found to be most important. In addition, as part of our regulatory analysis, we are performing a probabilistic risk assessment (PRA) focused on those initiating events which were judged to be most risk significant.

We recognize that following completion of our regulatory analysis, our quantitative understanding of shutdown risk will still be incomplete, and will not be comparable to the understanding gained from NUREG-1150 for power operation. You recommended that the staff perform a Level 3 shutdown PRA for the NUREG-1150 plants. While such a study may be desirable, we do not believe it is necessary in light of our ongoing regulatory analysis efforts. Furthermore, individual plant configurations during shutdown operation are more variable than at power. This not only complicates the performance of the risk study, it generally limits the generic applicability of the results. Therefore, we believe that a full Level 3 PRA of the NUREG-1150 plants is not warranted.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
SECY

ITEM B.4:

STATUS OF ACRS REVIEW OF THE NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL PHASE 2 STUDY REPORT ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

(DR. MILLER)

ITEM B.4: STATUS OF ACRS REVIEW OF THE NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL PHASE 2 STUDY REPORT ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the December 6, 1996 meeting with the Commissioners, the ACRS discussed the issues associated with the use of digital instrumentation and control (I&C) systems in nuclear power plants. The ACRS discussed the status of its review of the proposed update to Standard Review Plan (SRP), Chapter 7, "Instrumentation and Controls," Branch Technical Positions (BTPs), and regulatory guides for digital I&C systems. The ACRS also discussed the results of its review of the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 study report.

On October 13, 1995, the ACRS issued its report to the Commission regarding NAS/NRC Phase 1 study. In this report, the ACRS stated that the issues identified in the NAS/NRC Phase 1 study report will be important considerations as digital technology is used more extensively in nuclear power plants. The Phase 1 study report identified the following eight key issues - six technical and two strategic:

Technical

- software quality assurance
- common-mode software failure potential
- system aspects of digital I&C technology
- human factors and human-machine interfaces
- safety and reliability assessment methods
- dedication of commercial off-the-shelf hardware and software

Strategic

- case-by-case licensing
- adequacy of technical infrastructure

The NAS/NRC Phase 2 study report was completed on January 31, 1997. During its meeting of March 6-8, 1997, the ACRS discussed the Phase 2 study report with representatives of the NAS/NRC Committee and the Nuclear Regulatory Commission staff. In performing its Phase 2 study, the NAS/NRC Committee limited its work to those issues identified in the Phase 1 study. The specific charge for the Phase 2 study was as follows:

- Identify criteria for review and acceptance of digital I&C technology in both retrofitted reactors and new reactors of advanced designs.

- Characterize and evaluate alternative approaches to the certification or licensing of the digital I&C technology.

- Where sufficient scientific basis exists, recommend guidelines on the basis of which the Nuclear Regulatory Commission can regulate and certify (or license) digital I&C technology, including means for identifying and addressing new issues that may result from future development of this technology.

- Where sufficient scientific basis exists to make recommendations, suggest ways in which the Nuclear Regulatory Commission could acquire the required information.

The NAS/NRC Committee recommends that both the nuclear industry and the Nuclear Regulatory Commission be more proactive in participating in the relevant technical communities and strengthen infrastructure in digital I&C systems. The communication barriers among participants could be addressed systematically. The NAS/NRC Committee also noted that licensing criteria should be forged in a detailed interaction among the regulators, the industry, and the public. Generally, the NAS/NRC Committee emphasized the following:

- Deterministic assessment methods, including design basis accident analysis, hazard analysis, and other formal analysis procedures are applicable to digital systems, as long as they are applied with care.

- Software failure probability can be used for the purposes of performing PRA to determine the relative influence of digital system failures on the overall system. Including software failures in PRA is preferable to the alternative of ignoring software failures.

- Hardware and software must be treated together as a system; focusing solely on one or the other should be done with great caution.

- Most practical I&C systems cannot be exhaustively tested and thereby shown to be error free. However, adequate approaches exist and can be applied within practical resource constraints to support using digital systems in safety-critical applications in nuclear power plants.

The staff is in the process of incorporating the insights from the NAS/NRC Phase 2 study into the proposed final SRP, BTPs, and regulatory guides associated with digital I&C systems. The ACRS decided to postpone its comments on the NAS/NRC Phase 2 study report until after a meeting of its Subcommittee on I&C Systems and Computers, to be held on May 28-29, 1997, during which the Subcommittee will discuss the views of the Nuclear Regulatory Commission staff on the recommendations included in the NAS/NRC Phase 2 study report and other issues related to the Standard Review Plan (Chapter 7 update).

Attachments:

- Report dated October 13, 1995, from T. S. Kress, ACRS Chairman, to Shirley Ann Jackson, NRC Chairman, Subject: National Academy of Sciences/National Research Council Study on "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues" - Phase 1 (pp. 66-67)

- Letter dated October 31, 1995, from James M. Taylor, Executive Director for Operations, to T. S. Kress, ACRS Chairman, Subject: "The National Academy of Sciences' Report on Digital Instrumentation and Control, Safety and Reliability Issues" (p. 68)

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

October 13, 1995

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

Dear Chairman Jackson:

SUBJECT: NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY ON "DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS IN NUCLEAR POWER PLANTS, SAFETY AND RELIABILITY ISSUES" - PHASE 1

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 report on Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues. The NAS/NRC Committee Chairman described the results of the Phase 1 report. We also had the benefit of the documents referenced.

The objective of the Phase 1 study was to define the important safety and reliability issues concerning hardware, software, and human-machine interfaces that arise from the use of digital instrumentation and control technology in nuclear power plant operations. The report identifies eight key issues: six technical and two strategic. It notes that these issues are common to other industries where software is required for dependable operation of systems. The report succinctly presents the issues that the NAS/NRC Committee found to be important.

We agree that the issues identified in the Phase 1 report will be important considerations as digital technology is used more extensively in nuclear power plants. In the past, we have called attention to the effects of environmental stressors. The NAS/NRC Chairman stated that the NAS/NRC Committee considered, but decided not to raise this issue to the level of a "key technical issue." We continue to believe this is an important issue that the staff must address as it develops its regulatory guidance for digital systems. However, this is part of the broader issue of environmental qualification of safety-related equipment and does not need to be a key issue of the Phase 2 study.

We have concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. We believe it is important that the SRP and other regulatory guidance benefit from the insights in the Phase 2 report.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Report dated 1995, from the Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety, Board on Energy and Environmental Systems, Commission on Engineering and Technical Systems, National Research Council, Subject: Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues - Phase 1
2. Memorandum dated December 2, 1993, from Ivan Selin, Chairman, NRC, to NRC Commissioners, Subject: Computers in Nuclear Power Plant Operations
3. Letter dated July 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems
4. Letter dated August 23, 1994, from Ivan Selin, Chairman, NRC, to T. S. Kress, Chairman, ACRS, regarding ACRS letter of July 14, 1994 on National Academy of Sciences/National Research Council Proposal for a Study and Workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety"

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 31, 1995

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: THE NATIONAL ACADEMY OF SCIENCES' REPORT ON DIGITAL INSTRUMENTATION AND CONTROL, SAFETY AND RELIABILITY ISSUES

Dear Dr. Kress:

I am responding to your letter on this subject, dated October 13, 1995, in which the Advisory Committee on Reactor Safeguards (ACRS) commented on the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 Report, "Digital Instrumentation And Control Systems In Nuclear Power Plants, Safety And Reliability Issues."

We agree that the issue of environmental stressors is key to the qualification of safety-related digital instrumentation and control systems. Environmental stressors are defined as an issue in the NAS/NRC study Phase 1 report, but not as a "key technical issue." As you know from past briefings to the ACRS, the staff is conducting confirmatory research to investigate and characterize the failure modes and degradation mechanisms of digital technologies proposed for use in nuclear power plants. Furthermore, this research is assessing the impact of smoke on advanced instrumentation and control hardware in nuclear power plants. The goal of this research is to provide the technical basis for a regulatory guide on the environmental qualification of digital instrumentation and control systems. We informed the NAS/NRC Committee about our activities on this key issue in an October 17, 1995, meeting with them.

We share your concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. Our contract with NAS/NRC calls for the completion of the study by September 30, 1996, which includes the delivery of the Phase 2 report. The staff has expressed its concern and will continue to encourage a timely completion of the NAS/NRC study. We agree that it is important that the SRP and other regulatory guidance benefit from the insights expected from the Phase 2 report.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
SECY

ITEM B.5:

HUMAN PERFORMANCE PROGRAM PLAN

(DR. APOSTOLAKIS)

ITEM B.5: HUMAN PERFORMANCE PROGRAM PLAN

The NRC staff established a Human Factors Coordination Committee (HFCC) in 1994, which was comprised of representatives from the Offices of NRR, RES, NMSS and the Regions. The central task of the HFCC was to develop a Human Performance Program Plan (HPPP), which was initially issued in August 1995 and subsequently updated in 1996. The ACRS was first briefed on the HPPP during its February 1996 meeting. Subsequent meetings of the Human Factors Subcommittee were held in September and December 1996, to review the details of this Plan.

During its February 1997 meeting, the Committee completed its review of the HPPP and provided a report, dated February 13, 1997, to the Commission. In this report the Committee made several comments and recommendations, including the following:

- The HPPP is not a plan. It is, instead, an inventory of human performance projects within the agency. The HPPP should state explicitly what its goals are, what research efforts will be required to achieve these goals, and when and how it will be known that they have been achieved. The ownership of the present plan is diffuse. The success of such a plan as well as its dynamic nature require that ownership of the entire plan be clearly assigned.

- A well-planned research effort in human performance is urgently needed to support both the regulation of plant operations and the transition to risk-informed and performance-based regulation. The overall perspective that can be provided by high-level models of human performance would be helpful in the planning of this research effort. A number of such models are reviewed in NUREG/CR-6350, "A Technique for Human Error Analysis (ATHEANA)."

- The development of indicators of a good safety culture, the design of a meaningful human performance reporting system, and the impact of downsizing and deregulation on human performance should be major elements of the research effort. The human reliability analysis research project should also be part of the HPPP.

The Committee offered additional suggestions for improving the HPPP and expressed its intent to continue to work with the staff on this matter.

The Executive Director for Operations has responded to the above ACRS report in a letter dated April 10, 1997. His responses to the above three points can be paraphrased as follows:

- Agree that a comprehensive Human Performance Program Plan needs to be developed. The Office of Nuclear Regulatory Research will assume leadership for this effort. An agency-wide program Plan for human reliability assessment/performance evaluation will be developed for review by the end of June 1997.

- Agree that a comprehensive research plan is needed. Beyond providing a mechanism for coordinating current staff activities in these areas, the plan will articulate the conceptual relationships between human reliability analysis activities and those of human performance evaluation. In addition, the "ATHEANA" model, which was developed to improve human reliability analysis, will serve as the framework to guide activities associated with this initiative. Essential Plan elements will include: a mission statement, definition of strategies for achieving the mission, and identification of program areas to implement the strategies. ACRS will be briefed when the Plan is fully developed.

- Agree that research effort should include development of indicators of good safety culture, identification of the impact of downsizing and deregulation on human performance, and the design of a meaningful human performance reporting system. As part of its planned program to conduct operating events analysis to support human performance evaluation and human reliability analysis (HRA), the staff expects to recommend reporting requirements to better support HRA and human performance evaluation and to modify the LER coding scheme to include more human performance information.

The Committee plans to work with the staff in developing an effective Human Performance Program Plan.

Attachments:

- Report dated February 13, 1997, from R. L. Seale, ACRS Chairman, to Shirley Ann Jackson, NRC Chairman, Subject: Human Performance Program Plan (pp. 71-73)

- Letter dated April 10, 1997, from L. Joseph Callan, Executive Director for Operations, to R. L. Seale, ACRS Chairman, Subject: Human Performance Program Plan, with enclosure, Responses to ACRS' Conclusions and Recommendations (pp. 74-77)

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

February 13, 1997

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: HUMAN PERFORMANCE PROGRAM PLAN

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, we completed our review of the NRC activities identified in the Human Performance Program Plan (HPPP). Our Subcommittee on Human Factors met on September 20 and December 3, 1996, to review these activities. During these reviews, we had the benefit of discussions with representatives of the staff.

In your remarks of December 2, 1996, to all NRC employees, you stated:

> As we move to an era of nuclear power industry restructuring and declining NRC and industry resources, it is imperative that we are able to diagnose potentially declining licensee performance as early as possible.

We agree with your assessment. We believe that an appropriate HPPP would contribute significantly to the development of such diagnostic tools.

Conclusions and Recommendations

1. The HPPP is not a plan. It is, instead, an inventory of human performance projects within the agency. The HPPP should state explicitly what its goals are, what research efforts will be required to achieve these goals, and when and how it will be known that they have been achieved. The ownership of the present plan is diffuse. The success of such a plan as well as its dynamic nature require that ownership of the entire plan be clearly assigned.

2. A well-planned research effort in human performance is urgently needed to support both the regulation of plant operations and the transition to risk-informed and

performance-based regulation. The overall perspective that can be provided by high-level models of human performance would be helpful in the planning of this research effort. A number of such models are reviewed in NUREG/CR-6350.

3. The development of indicators of a good safety culture, the design of a meaningful human performance reporting system, and the impact of downsizing and deregulation on human performance should be major elements of the research effort.

Discussion

Operational experience has shown that human performance is a major factor in the safe operation of nuclear power plants. Understanding what can go wrong at a plant requires an integrated evaluation of both hardware and human performance; i.e., the plant must be viewed as a sociotechnical system. In particular, the term "human error," which carries the implication that the operators are to be blamed, is inaccurate in many instances and one must investigate and understand the context within which plant personnel function. This context is determined by both the design and the physical conditions of the plant, as well as by the prevailing safety culture.

The development of a plan for research on human factors is certainly not a simple task. This task would be made easier and the recommendations more convincing if the task were guided by a high-level model that identifies the important elements that influence the likelihood of unsafe human acts. Various models and taxonomies have been proposed in the literature and some are beginning to receive wide acceptance. Human performance models and error classifications that could be suitable guides for developing a research plan are being used in other projects in the Office of Nuclear Regulatory Research. The models discussed in NUREG/CR-6350, along with insights from operational experience, could serve to guide the development of an HPPP.

One specific element we would like to see addressed in the HPPP is the impact of situational assessment on compliance with procedures. Investigations of actual incidents and simulator exercises from nuclear and other industries have demonstrated the importance of what Professor James Reason of the University of Manchester calls "intended violations" (circumventions) of procedures by plant personnel. The researchers who collected data from simulator exercises point out that these were not necessarily errors; the operators simply did what they felt was the optimal response to the evolving accident. We believe there is a need to understand the reasons for such deviations and how training, procedures, and the plant safety culture could be modified to eliminate "circumventions" to the extent possible.

The present HPPP contains elements that are worth pursuing. Other elements that should be contained in the HPPP include activities to gain a better understanding of the concept of safety culture and to develop indicators of a good safety culture. The human reliability analysis research project should also be part of the HPPP. We will continue to work with the staff in developing an effective HPPP.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Memorandum dated July 31, 1996, from Cecil Thomas, Office of Nuclear Reactor Regulation, to John Larkins, ACRS Executive Director, Subject: Forwarding Human Performance Plan Rev. 1
2. Office for Analysis and Evaluation of Operational Data Report E-95-01, "Operating Events with Inappropriate Bypass or Defeat of Engineered Safety Features," July 1995
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6093, "An Analysis of Operational Experience During LP&S and a Plan for Addressing Human Reliability Issues," June 1994
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6265, "Multidisciplinary Framework for Analyzing Errors of Commission and Dependencies in Human Reliability Analysis," August 1995
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6350, "A Technique for Human Error Analysis (ATHEANA)," May 1996
6. Reason, J.T., Human Error, Cambridge University Press, Cambridge, United Kingdom, 1990
7. R. Montmayeul, F. Mosneron-Dupin, and M. Llory, "The Managerial Dilemma between the Prescribed Task and the Real Activity of Operators: Some Trends for Research on Human Factors," Reliability Engineering and System Safety, 45:67-73, 1994
8. U. S. Nuclear Regulatory Commission, NUREG/CR-6208, "An Empirical Investigation of Operator Performance in Cognitively Demanding Simulated Emergencies," July 1994
9. International Atomic Energy Agency, Vienna, International Nuclear Safety Advisory Group, "Safety Culture," Report 75-INSAG-4, 1991
10. NRC Chairman Shirley Ann Jackson's remarks to all NRC employees, December 2, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 10, 1997

Dr. Robert L. Seale, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Seale:

SUBJECT: HUMAN PERFORMANCE PROGRAM PLAN

This responds to your letter dated February 13, 1997, in which you provided comments concerning the Human Performance Program Plan. Your conclusions and recommendations and the staff's responses to each are enclosed.

Sincerely,

L. Joseph Callan
Executive Director
for Operations

Enclosure: As stated

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY
OGC

Enclosure

Responses to ACRS' Conclusions and Recommendations

1. The HPPP is not a plan. It is, instead, an inventory of human performance projects within the agency. The HPPP should state explicitly what its goals are, what research efforts will be required to achieve these goals, and when and how it will be known that they have been achieved. The ownership of the present plan is diffuse. The success of such a plan as well as its dynamic nature require that ownership of the entire plan be clearly assigned.

The ACRS is quite correct. As stated in our February 4, 1997, responses to ACRS questions, the HPPP was intended to function as a mechanism to coordinate human factors activities among the Agency's offices. However, we agree that a comprehensive program plan needs to be developed.

Presently, ownership for coordination for the HPPP is in NRR. Since future needed efforts are developmental and are expected to involve comfirmatory research, leadership will be shifted to the Office of Nuclear Regulatory Research. An agency-wide program plan for human reliability assessment and human performance evaluation is expected to be developed for review by the end of June.

2. A well-planned research effort in human performance is urgently needed to support both the regulation of plant operations and the transition to risk-informed and performance-based regulation. The overall perspective that can be provided by high-level models of human performance would be helpful in the planning of this research effort. A number of such models are reviewed in NUREG/CR-6350.

We agree that a comprehensive research plan is needed. Consistent with the ACRS's suggestion, the staff is developing a plan for integrating activities in human reliability assessment and human performance evaluation. Beyond providing a mechanism for coordinating current staff activities in these areas, the plan will articulate the conceptual relationships between human reliability analysis activities and those of human performance evaluation. In addition, the "ATHEANA" model, which was developed to improve human reliability analysis, will serve as the framework to guide activities associated with this initiative.

The essential elements of the plan will be a statement of the mission of the human reliability assessment and human performance evaluation initiative in the Agency, a definition of strategies for achieving the mission, and the identification of program areas established to implement the strategies. Within each program area, summaries of ongoing and planned activities will describe their relevance and consistency with the defined mission, strategies, and programs. Such an approach to planning is expected to better highlight any redundancy, gaps, and disproportionate emphasis in current and proposed staff efforts. We will brief the ACRS on the program plan after it is fully developed.

3. The development of indicators of a good safety culture, the design of a meaningful human performance reporting system, and the impact of downsizing and deregulation on human performance should be major elements of the research effort.

We agree that the research effort should include development of indicators of good safety culture, identification of the impact of downsizing and deregulation on human performance, and the design of a meaningful human performance reporting system.

As part of its planned program to develop the technical basis and guidance on management and organizational influences in human performance and plant risk, the influences of management practices and safety culture on human performance and human reliability will be identified. Additionally, the impact of downsizing and deregulation on human performance and human reliability will be investigated. As part of efforts to improve the Senior Management Meeting process, the staff is identifying measures of economic stress that can be used to identify plants for increased safety monitoring.

Licensees are required to report data to the NRC on factors that influence operating events, including factors that contribute to human performance during events. However, the human performance data collected are not always sufficiently detailed to provide the basis for formulating research or regulatory programs. As part of its planned program to conduct operating events analysis to support human performance evaluation and human reliability analysis, the staff expects to recommend reporting requirements to better support HRA and human performance evaluation and to modify the LER coding scheme to include more human performance information.

ITEM B.6:

ACRS REPORT TO CONGRESS ON NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

(DR. POWERS)

ITEM B.6: ACRS REPORT TO CONGRESS ON NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

The Advisory Committee on Reactor Safeguards is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209 to report to Congress each year on the Safety Research Program of the U.S. Nuclear Regulatory Commission.

On February 21, 1997, the ACRS provided its 1996 report to the Congress. In this report, the ACRS expressed its views on the potential impact of a reduced Safety Research Program on regulatory reform and the ability to provide adequate safety oversight for a changing nuclear industry.

The highlights of the ACRS report include the following:

- Continued availability of unbiased safety research information will be essential as the NRC establishes itself as the leader in the national effort to reform the regulatory process to focus on real risks, continued safety of operating nuclear power plants, and the performance of licensees.

- The Safety Research Program has enabled the NRC to develop a probabilistic risk assessment method that can provide quantitative measures of the real risks of nuclear power.

- Based on information that has come from the Safety Research Program, operational experience, and the ability to quantify risk, the NRC has been able set forth safety goals that define how safe is safe enough.

- Understanding of risk has reached the point that it can be used to reformulate the regulatory structure to focus both licensee and regulatory attentions on what is significant to safety in a cost-effective way.

- The Safety Research Program has aided the NRC in the development of standards for the use of risk assessment in the regulatory process.

- A risk-informed regulatory structure will be essential for the NRC to respond to changes taking place within the nuclear industry in response to economic pressures. These changes include modernization of instrumentation and control systems, downsizing workforces, and extending fuel lifetimes. Each of these changes mandates evolution of NRC rules and regulations to assure continued, adequate protection of the public health and safety.

- Funding for the Safety Research Program has been reduced to a level that may not allow a cost-effective response by NRC to the new challenges.

- The Safety Research Program will have to be sustained and even augmented if the NRC is to complete its transformation to risk-informed and performance-based regulatory approach. The NRC effort to establish a risk-informed and performance-based regulatory approach is an example for other regulatory agencies that should not be allowed to fail.

- Without the needed research support, the NRC may be forced to rely on historical, conservative, costly regulations not necessarily focused on risks.

The ACRS plans to continue its review of various elements of the NRC Safety Research Program.

Attachment:

- Report dated February 21, 1997, from R. L. Seale, ACRS Chairman, to the Honorable Albert Gore, Jr., President of the United States Senate, and to the Honorable Newt Gingrich, Speaker of the United States House of Representatives, Subject: The Advisory Committee on Reactor Safeguards Report on Nuclear Safety Research and Regulatory Reform (pp. 80-86)

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

February 21, 1997

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

Dear Mr. President:

I am pleased to transmit to the Congress the 1996 report of the Advisory Committee on Reactor Safeguards on the U. S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

Enclosure:
U. S. Nuclear Regulatory Commission, "The Advisory Committee on Reactor Safeguards Report on Nuclear Safety Research and Regulatory Reform," dated February 1997

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

February 21, 1997

The Honorable Newt Gingrich
Speaker of the United States
House of Representatives
Washington, D.C. 20515

Dear Mr. Speaker:

I am pleased to transmit to the Congress the 1996 report of the Advisory Committee on Reactor Safeguards on the U. S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

Enclosure:
U. S. Nuclear Regulatory Commission, "The Advisory Committee on Reactor Safeguards Report on Nuclear Safety Research and Regulatory Reform," dated February 1997

A REPORT TO THE CONGRESS
OF THE UNITED STATES OF AMERICA

ON

NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

BY

THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U.S. NUCLEAR REGULATORY COMMISSION

FEBRUARY 1997

THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

The Advisory Committee on Reactor Safeguards, in the past, reported on very specific reactor safety research issues and programs. In light of the diminished resources available to support the U.S. Nuclear Regulatory Commission's Safety Research Program, we have chosen, instead, to report on the potential effects of a reduced Safety Research Program on regulatory reform and the ability to provide adequate safety oversight for a changing nuclear industry.

A vigorous research program dealing with the safety of commercial nuclear power production has served the Nuclear Regulatory Commission and the public well in the past. The continued availability of unbiased safety research information will be essential as the Nuclear Regulatory Commission establishes itself as the leader in the national effort to reform the regulatory process to focus on real risks, continued safety of operating nuclear power plants, and the performance of licensees. At the same time, initiatives taken by the commercial nuclear power industry in response to ongoing and anticipated deregulation of electrical power generation make it even more important that the Nuclear Regulatory Commission continue to have a Safety Research Program that provides the information needed to modify and improve its regulations to protect public health and safety.

From the inception of the civilian use of nuclear energy to generate electrical power, public safety has been of paramount concern. Initially, little experience and few industrial safety standards were available to ensure that nuclear power could be generated safely. As a result, prescriptive, highly conservative approaches that blanketed all aspects of nuclear power generation

were adopted by both the regulatory authority and the industry. Faults and vulnerabilities identified through operation of nuclear power plants were used to add layers of protection on this regulatory structure. Indeed, regulation of nuclear power generation has been successful in protecting public safety in this country. But, safety has been achieved through highly conservative regulation at great cost to both the producers and consumers of nuclear power.

As nuclear power generation has matured, experience has been gained in our understanding of the real risks of nuclear power. The Safety Research Program has enabled the Nuclear Regulatory Commission to develop a method called probabilistic risk assessment that can provide quantitative measures of these risks. The sophistication of this understanding has reached the point that it is now possible to initiate a reformation of the regulatory structure for nuclear power generation. This reformation will focus attention on what is significant to safety and at the same time will allow the industry to identify and use cost-effective strategies to mitigate risks. Reformation of regulation of all types to focus on risk is, of course, a national priority. The Nuclear Regulatory Commission is taking the lead in this national effort with its policy of risk-informed and performance-based regulation. Based on information that has come from the Safety Research Program, operational experience, and the ability to quantify risk, the Nuclear Regulatory Commission has been able to set forth safety goals that define how safe is safe enough. By working with individuals experienced in plant operations and using the tools of risk analysis, the NRC can now identify regulations that do not contribute to safety, and it will be able to define a rational, cost-benefit basis for imposing additional regulatory requirements.

Steps are being taken in the direction of risk-informed and performance-based regulation. The performance-based maintenance rule (10 CFR 50.65) is a tangible accomplishment. Rather than imposing bureaucratic prescriptions on every aspect of safety system maintenance, this rule allows the industry to find creative strategies to meet performance objectives approved by the Nuclear Regulatory Commission based on risk information. Satisfactory performance by licensees is rewarded by reductions in regulatory burdens while performance failures elicit increased regulatory scrutiny.

The Safety Research Program has aided the Nuclear Regulatory Commission in the development of standards for regulatory use of risk assessment. This would permit additional uses of this approach to focus dwindling resources on issues of most importance for protecting public health and safety. Target applications of these new standards are in-service inspection, in-service testing, and technical specifications for reactor safety systems. Continued research will be essential for further regulatory reforms.

New challenges to the regulation of nuclear power are emerging. These challenges come from the deregulation of electrical energy production and the need for the nuclear power industry to become more cost competitive. The nuclear industry is aggressively pursuing changes to remain economically viable. These changes could have significant safety implications that will require regulatory approval when they affect the licensing basis for nuclear power plants. Among the changes under consideration are increased fuel lifetimes, elevated operating power, digital instrumentation and control systems, and downsized work forces. Each of these changes could challenge the existing regulations for the protection of public health and safety. We believe that applied regulatory research programs will be required to develop bases/criteria for regulatory approval of these changes. Of

particular importance are the changes that may affect human performance in the operation of nuclear power plants.

Funding for research activities has fallen by a factor of about 3 over the last 10 years and all evidence points toward continued reductions in the future. While much of this decrease can be attributed to the maturation of the technology, funding for the Safety Research Program has been reduced to a level that may not allow a cost-effective response to these new challenges. The Nuclear Regulatory Commission now does not have the technical tools needed to evaluate all of the safety implications of extending fuel lifetimes to the extent the nuclear industry has requested. It cannot evaluate quantitatively the risk implications of personnel reductions and modernization that are being proposed by the nuclear industry. The Safety Research Program will have to be sustained and even augmented if the Nuclear Regulatory Commission is to complete its transformation to risk-informed and performance-based regulatory approach. Without the needed research support, the Nuclear Regulatory Commission may be forced to rely on historical, conservative, costly regulations not necessarily focused on risks. Safety innovations by the industry may be stifled. The opportunity to use regulation of nuclear power as an example of successful regulatory reform may be lost.

ORIGINAL

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Title: MEETING WITH NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC) - PUBLIC MEETING

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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MEETING WITH NUCLEAR SAFETY

RESEARCH REVIEW COMMITTEE (NSRRC)

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PUBLIC MEETING

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Nuclear Regulatory Commission

Commission Hearing Room

11555 Rockville Pike

Rockville, Maryland

Friday, May 2, 1997

The Commission met in open session, pursuant to notice, at 10:53 a.m., the Honorable SHIRLEY A. JACKSON, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

SHIRLEY A. JACKSON, Chairman of the Commission

KENNETH C. ROGERS, Member of the Commission

EDWARD McGAFFIGAN, JR., Commissioner.

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

JOHN C. HOYLE, Secretary
KAREN D. CYR, General Counsel
E.T. BOULETTE, NSRRC Chairman
S. GEORGE BANKOFF, NSRRC
MICHAEL W. GOLAY, NSRRC
CHARLES MAYO, NSRRC
CHRISTINE M. MITCHELL, NSRRC
JOHN TAYLOR, NSRRC
SUMIO YUKAWA, NSRRC
DAVID MORRISON, Director, Office of Nuclear Regulatory Research

P R O C E E D I N G S

[10:53 a.m.]

CHAIRMAN JACKSON: Good morning, ladies and gentlemen.

I am pleased to welcome Dr. E. Thomas Boulette and members of the Nuclear Safety Research Review Committee, and Dr. David Morrison, Director of the Office of Nuclear Regulatory Research, to brief the Commission on recent activities of the committee.

The Nuclear Safety Research Review Committee or the NSRRC, as it is called, advises the director of Nuclear Regulatory Research and, through him, the Commission on the quality and conduct of NRC research activities and gives recommendations concerning the overall management and direction of the Nuclear Safety Research Program.

At today's briefing, the following topics will be discussed. First, observation and recommendations of four subcommittees, among them the Materials and Engineering Subcommittee, a joint report from the INC and Human Factors Subcommittee and the PRA Subcommittee. And, finally, the Accident Analysis Subcommittee.

Also discussed will be research core capabilities and the committee's view of these, comments on the committee's effectiveness in support of research and comments addressing the Commission's questions concerning

human reliability analysis and their relationship to PRA.

The Commission appreciates your effort and look forward to hearing from you. I understand that if there is any presentational material, it has already been made available.

Please start, Mr. Boulette.

DR. BOULETTE: Thank you, Chairman Jackson, and good morning, Commissioner Rogers, Commissioner McGaffigan.

We are pleased to be here this morning to give you our views of the research program that the NRC is very dependent upon. We will also afford you an opportunity to meet the membership. I know you haven't done that before. And in fact what we have planned is that every member at the table be speaking.

The agenda is relatively tight. I was present for the ACRS meeting and I can see how these proceedings go. I am going to encourage the membership of this committee to be cognizant of the time and the messages that we are trying to present to you.

CHAIRMAN JACKSON: It is we who caused the delay.

DR. BOULETTE: A couple of comments about the committee itself.

Historically, there have been 12 members on this committee. Currently there are only seven. Soon, there will be only six unless we -- unless Dr. Morrison is

successful in recruiting some other members. One of the concerns that we have is the breadth of expertise that the Committee is trying to sustain so we will be working on that over the next couple of months.

To make the committee effective, we have broken it up into five subcommittees, four of which are very active. One is somewhat inactive because of the area of expertise on high-level waste.

The four committees will report to you this morning their findings at their recent meetings and their views of the specific areas of which they have responsibility.

The committees include the committee on PRA. Mike Golay is the chair of that committee. Another committee is Human Factors and INC. Charles Mayo is the chair of that subcommittee. Accident Analysis is the third subcommittee and George Bankoff is the chair of that subcommittee. And Materials and Engineering is the fourth and that is chaired by Sumio Yukawa.

We try to meet twice a year as the full committee and the subcommittees try to meet two to three times a year. We have no staff so most of what you get is a bit sophomoric, I think, in terms of the quality of the typing. That is because I do the typing of the reports.

We have tried to address the concerns that the

Commission had in terms of the interface between this committee and the ACRS. I think we have been very active this past 12 months in doing that. The way we do that is to try to be very cognizant of their schedule and the meetings that they have and then selectively select a member of our committee to attend some of the meetings. There have been at least a half a dozen or so meetings that we have been participating in. It has been very useful to us. It helps to focus on what we may want to talk to the staff about.

With these preliminary comments, I will move on to the next subject on the agenda which is a report on the joint meeting of the INC and Human Factors and the PRA subcommittees and that is Charles Mayo.

MR. MAYO: Okay, thank you.

Our committees had a joint meeting primarily to review and prepare response to the questions that had been posed about the use of human reliability analysis and PRA and we reviewed the human performance program plan, the PRA implementation plan and other material provided to us and concluded that the research projects in human factors and human reliability analysis are largely unrelated. This seems to be primarily driven by user needs to perform reliability analysis and the licensing space as opposed to developing methods and data specifically directed to the human reliability analysis problem and applications of it.

RES does have two programs in human reliability analysis area and considers them to have somewhat limited scope. There is the Athena project on the areas of the commission and the organizational factors management. I would have to say that our subcommittee has been concerned about the issue of the organizational factors research program for a number of years and I came on the committee as previous work was ending so I don't know the historical details but we still have some concerns about progress in that area.

Additionally, in looking at the programs that were going on or could be going on, the data needs, we had the analysis that there was likely to be significant relevant experience in the NRC operating database and we could see references in some of the program plans to this being collected and the licensee event report improved and so on to develop for human reliability data. We feel that this is a research area or opportunity for data that should not be ignored and particularly in comparison to the classical human reliability analysis type data that has come from other industries.

And the final point was there was a belief that the two projects that are currently going on did not constitute a developed research plan to develop the human reliability, human factors analysis into use in the PRA but

that improvements certainly could be made through a longer term program.

CHAIRMAN JACKSON: Let me ask you a couple questions.

You talked about the human reliability analysis program having limited scope. Has the committee made any specific recommendations on an expanded scope?

MR. MAYO: We have not had the opportunity to do that. We had a busy meeting when we got to this point.

CHAIRMAN JACKSON: Do you plan to make recommendations?

MR. MAYO: We are trying to get together again in the early part of the summer and discuss this, after we have had a better sense for material we received and feedback from the ACRS.

CHAIRMAN JACKSON: You mentioned that user needs do not address development of human reliability analysis, that portion of the PRAs -- I'm going to call it HRAs from now on. To what extent has the current state of the art in HRA limited our ability to apply PRA results in the regulatory arena.

MR. MAYO: That question I must defer to some of my colleagues on the Committee.

DR. BOULETTE: Christine, can you take that question?

To what extent can HRA be effectively used in PRA, I think, is the nature of the question.

CHAIRMAN JACKSON: Yes, in the regulatory arena.

MS. MITCHELL: I think that if you have a -- to the extent that you are able to model human operators, you have a stronger model. To the extent that you are not or you don't have particularly valid data for that, it limits, limits your overall model.

My understanding is that HRA is pretty primitive at this point in time. My understanding from your last session with Dr. Apostolakis is that -- and I concur -- is that it's a mess. So it needs some attention, although I caution that this isn't just a matter of money and effort; this is the state of affairs in lots of other industries. Modeling human performance and using those models in an analytic way is not widely done anyplace.

CHAIRMAN JACKSON: And you mentioned that analysis of operating experience should be a resource for relating HF, HRA and PRA. Why is that not happening?

MR. MAYO: I believe it is happening in certain ways. Our exposure to date has been limited to what we read about projects in the program plans, particularly in AEOD activities, which we haven't gotten into much detail on.

I guess our concern was the absence of seeing active work going on within the RES division itself.

MR. GOLAY: I think there is another point, if I can offer a comment, which is that in order for data to be useful in modeling, there has to be a coupling between the model development and understanding that the case is being analyzed. And the lack of interaction between research and AEOD was effectively a lost opportunity that we were drawing attention to, in that AEOD has been using PRA to try to understand some events, precursor analysis, for example. But the feedback link to the research program and to setting the agenda to refining the models to understanding results that they are getting wasn't there.

So the format, for example, in which the AEOD evidence was being interpreted was not in a state where researchers could make easy use of it so it was not making the kind of contribution that could be made at fairly modest marginal cost, it appeared.

CHAIRMAN JACKSON: Well, in fact, now AEOD and research are part of the same organization and it was meant to address some of this. So, Dr. Morrison, can you give is some edification relative to what is happening in this regard?

DR. MORRISON: Yes. We are very, very much moving out based upon both the recommendations that ACRS made in this broad area as well as the comments that NRC has made.

Two things to note, one is that there has been a

recent reorganization within the Office of Research that placed HRA or actually people from PRA that had been doing some HRA activities, into the human factors area and vice versa, so that they are closely coupled and, second, this group is developing a human performance, human reliability plan that is basically going to be an agency wide plan that has its origins in research and trying to address the immediate needs that have been raised by both committees.

That plan should be available for review, I would think, by the subcommittee here at the early summer meeting so that there will be an opportunity to get feedback on the plan that is being developed.

CHAIRMAN JACKSON: What about the issue of specifically linkages between research and AEOD or research drawing on the AEOD operating database?

DR. MORRISON: Well, those have existed in the past. Obviously, they need to be strengthened. They are in the process of being strengthened. We have been working quite closely with AEOD in the accident sequence precursor efforts and we can broaden out on that particular basis.

CHAIRMAN JACKSON: Well, the question I guess I am asking you is, as part of this agency wide plan, is this issue of cross linkage and, you know, use of the database being explicitly addressed? Because you are right, it has existed all the time but the Committee is making a statement

as has made -- been made by ACRS that the activities are unrelated and that the database has not been drawn upon.

DR. MORRISON: Well, it will be explicitly addressed in the plan and what steps we will take to make sure that that continues.

CHAIRMAN JACKSON: Commissioner Rogers?

COMMISSIONER ROGERS: No questions.

COMMISSIONER McGAFFIGAN: Could I ask the relationship between your body and ACRS in looking at this issue? You are both looking at it simultaneously and reviewing plans, both finding them not very acceptable at the moment and telling the staff that they have to rework, as ACRS, Mr. Apostolakis, said, the staff is in agreement and has gone back to the drawing board.

But what is the value added of your look at it compared to ACRS or how should we think about rationalizing that?

MR. MAYO: Well, we are developing a relationship with ACRS. In my particular case, I was unable to attend their last subcommittee meeting so I personally did not participate, but our other committee members have been attending the ACRS meetings and I have seen, as mentioned earlier, progress in coordinating our activities.

MR. GOLAY: I will add one thing.

The mandates of the two groups are somewhat

different, in that our committee is concerned with the research program throughout NRC. The ACRS is concerned with the reactors, reactor-related activities of NRC and there is an intersection concerned with research related to reactors, which is the bulk of research but not entirely.

CHAIRMAN JACKSON: Okay.

DR. BOULETTE: The next subject that we wanted to discuss with you is the subject of PRA and its use in risk-informed performance-based regulations and Mike Golay will speak to that.

MR. GOLAY: The subcommittee we have on PRA has put together partly to help the research group develop the capabilities which are needed to support all of the NRC in making performance-based regulation an effective reality. So I will make my comments sort of from that perspective.

Whenever we have reviewed their programs, it has always been to try to answer questions about what do they need to do in order to be an effective support and the thing that we are seeing is that there are ways that research could be much more valuable, primarily in promoting fluency concerning PRAs throughout the agency. They participate with AEOD in training and one of the things which we can see is that sensitivity to what PRA will tell you really has not permeated very much in the functioning of the agency, at least anecdotally it appears that way when you talk to

licensees and ask, do you ever see any evidence that performance-based regulation is a reality within the agency or in terms of how you resolve issues in dealing with the NRC and the answer is consistently that there is --

CHAIRMAN JACKSON: Are you saying risk-informed performance-based regulation or are you saying performance-based regulation?

MR. GOLAY: I mean the former. I was trying to be brief.

CHAIRMAN JACKSON: Okay, I just want to be sure I understand what you are talking about.

MR. GOLAY: No, that's what I mean.

That one of the things they say is that the staff appear really not to be knowledgeable about PRA or even aware that it is one of the tools which could be used in dealing with the questions which come up with the licensees.

CHAIRMAN JACKSON: Now, they are talking about the staff lacking knowledge, are you talking at the level of the resident inspectors, at the region-based inspectors?

MR. GOLAY: It is at the regions primarily, that's right. So consistently when you ask them, well, are you trying to pose some of your arguments in risk-based terms, they say, no, because the NRC is unable or unwilling to communicate in those terms.

Which comes back then to the research program

because of the role that they play in instilling those capabilities. I think one message is that they could be very valuable in being more vigorous in this kind of thing so that if you look at the second bullet, when we say, what is really meant here, say greater use of PRA is needed in guiding regulation, it really means in terms of dealing with licensees as opposed to formulation of policy or determination of new regulatory statements.

CHAIRMAN JACKSON: So this committee, has this committee had any role in reviewing or participating in the review of the PRA reg guide or standard review plan?

MR. GOLAY: Only because I took the initiative to get those documents and review them. Had I not done so, they would not have come to our attention.

CHAIRMAN JACKSON: You provided commentary back to the staff?

MR. GOLAY: No, I read them so I could, first of all, know what they are trying to do and if I were asked anything about them have some kind of answer.

But I am saying routinely that kind of thing is not brought to our attention.

CHAIRMAN JACKSON: I see.

MR. GOLAY: As we have been working so far.

CHAIRMAN JACKSON: So research has not had a role in reviewing these documents themselves?

DR. MORRISON: Well, research has had an integral role in developing the --

CHAIRMAN JACKSON: Developing them, right. But this committee was not asked to review them.

DR. MORRISON: No. This committee generally has not been asked to review regulatory guides or anything related to the rulemaking process.

CHAIRMAN JACKSON: Okay.

MR. GOLAY: Right.

The third bullet goes to also the interaction with the licensees and within the staff in that the other thing that I at least have become aware of is with the two thrusts that are going on in the agency at the moment, one concerned with strict conformance to commitments that licensees have made, that there is effectively an interference that is being created which I would say is working against performance-based regulation in that the licensees are asking, well, should we be paying attention to the letter of the law in fine detail without regard to the substance of what is being regulated and I think they are concluding that, yes, that that is the case, at least in the past year or so.

And, contrasting that to, well, should I try to use risk-informed performance-based regulatory approaches to problems which are, as you know, concerned with the

substance. There is some variation in how you pose those arguments.

Uniformly, what I am seeing is they are basically ruling out performance-based regulation as an approach and this has, I think, an important effect because it also decreases the resources within the licensees to play ball in the risk-informed performance-based regulatory arena. So we have got sort of systematic interaction here, which is undermining the needed growth of capabilities to support that approach to regulation, both within the utilities and, I would say, within the NRC.

You know, George, in the last session, spoke about this maturation time which is needed before the licensees are able to actually use this way of approaching problems effectively and what I am observing is that in fact that maturation is being suppressed by these two parallel sort of conflicting messages.

CHAIRMAN JACKSON: So you are suggesting that the licensees are suggesting that they should be relieved from their commitments because they have no safety significance?

MR. GOLAY: Not at all. I would say it is a matter of style rather than whether they feel they need to be strongly committed because resources have to be divided in some fashion and what they are doing is putting their resources into compliance and they are taking them away from

building the capability for risk-informed performance-based regulation. So it is having an effect in that fashion.

CHAIRMAN JACKSON: Well, you know, we have a risk-informed enforcement policy and we are a regulatory agency and so I think, you know, we have to come around this issue of compliance issues versus safety issues. If, in fact, licensees feel that there are compliance issues that do not have a safety basis, I think all of us would welcome them being brought to our attention because I think we are not interested in having compliance against things that do not have a safety case. But I think that you cannot talk about a regulatory agency not expecting people to comply with something.

MR. GOLAY: Absolutely. Absolutely.

CHAIRMAN JACKSON: Okay.

MR. GOLAY: No question.

CHAIRMAN JACKSON: So I don't think we want to get off into those kinds of pejorative discussions.

MR. GOLAY: That's right. I only wanted to draw attention to some interactions which are effecting the advancement of performance-based regulation, which I think is really one of the key contributions that the agency has been making in recent years to improving safety.

We spoke -- on the fourth bullet, we already spoke about the coupling between AEOD and research and so I don't

think we need to say more about that. And I would say that basically the agency really can be congratulated for making good progress in development of the draft reg guides, revisions to the standard review plan, development of some PRA tools like the Saphire code suite.

So in building this infrastructure, there are some good things, good things to point to, and there are other areas where, if the resources could be applied, it would be good to make more rapid progress. I would say these primarily concern dealing with uncertainty that was spoken about in the last session.

I would say, dealing with data was not talked about very much but, again, this is an area where the NRC and particularly research could be effective in that what you really need is a systematic method for collecting data in a format which is going to be easily scrutinized, permit the data to be scrutinized and transformed into a format that will be useful in PRAs, and right now we don't have that. What we have is a more of an anecdotal data collection system existing within NRC, in INPO, in EPRI with the various PRA vendors and so on. So standardization and attention to that is very important because collection of data is a long-term process but a little up-front investment can pay off by being made early.

CHAIRMAN JACKSON: Do we need the reliability data

rule?

MR. GOLAY: I don't know the answer. I am more comfortable stating the goal than addressing the tactic.

CHAIRMAN JACKSON: Well, I think we all have the same goal. I think to get there requires a tactic.

MR. GOLAY: Yes. But in addressing sort of agenda items where research might think about applying more resources, those are I would say the two primary ones.

CHAIRMAN JACKSON: Okay.

Commissioner Rogers?

COMMISSIONER ROGERS: Yes. We are talking here about PRA and we are also talking about risk-informed performance-based regulation and the point that I feel sometimes gets lost here is the value of risk, a risk-informed point of view that is not entirely based upon a full quantitative PRA but it is, nevertheless, a risk ranking, a risk assessment in some way that isn't dependent totally upon having data, reliability data, that just simply may not exist. And yet that perspective is a very valuable one.

I just wonder what your thoughts are on that, because it seems to me that we tend to keep coupling PRA or interpreting risk-informed performance-based regulation or risk-informed regulation in any way, whether it is performance-based or not, on the notion that it starts with

a PRA. It doesn't have to depend totally on a full PRA. A risk assessment can still be a very valuable beginning point for looking at a system and that is happening in the materials area but -- and I wonder to what extent you are aware of that.

In the materials processing plants, that is exactly what they are doing. They are not doing PRAs but they are doing risk categorization and risk classification as part of their overall systems analysis.

DR. YUKAWA: I would just like to make a comment here that I am a member of the PRA subcommittee but also I am making this comment as a member of the AMSE Boiler and Pressure Vessel Code at ISI. There have passed now in Section 11 several risk-based inspections. They are on piping. And I think the industry will look to what the Commission will do about that to see what the future holds for them. So that should be coming through as a code case pretty soon.

CHAIRMAN JACKSON: Within months or this year?

DR. YUKAWA: It has passed all the main committees now so it should be coming up within the next, latter half of this year anyway.

CHAIRMAN JACKSON: So this is specifically with reference to piping?

DR. YUKAWA: This is for -- there are two kinds of

code cases. One is very specific to a very specific line, a pipeline. The other is a more general one about risk-based inspection for a larger category of pipes. The first, more restrictive one, is only for class one piping.

CHAIRMAN JACKSON: Okay.

Commissioner McGaffigan? Okay.

DR. BOULETTE: The next item on the agenda is going to be discussed by Christine Mitchell. The subject is her review of the National Academy of Sciences report on digital INC.

Christine.

MS. MITCHELL: Thank you.

I guess I should introduce this by saying it is not really a review because I wear two hats. I served on that National Academy committee as well as on the NSRRC and so what is on your handout is just a high-level set of points and I would be happy to field questions.

I think the major things that the National Academy report provided include an affirmation that although digital technology is state-of-the-art technology and continues to change at an increasing rate, there is a great deal of experience with digital technology both in the nuclear industry and in many other industries. The point being that there is a tremendous amount of experience out there, even though it is not necessarily U.S. safety system experience

in the nuclear industry.

The second is that the committee affirmed that digital INC has the potential to enhance safety and reliability so we agree that this is a productive avenue to pursue, basically agreeing with agencies such as the FAA, both on the flight deck and in air traffic control, that digital technology can make an improvement as well as being a cost efficient way to go.

And finally, in terms of nuclear applications and their particular cultural history and movement from analog to digital technology, that there are some special concerns that need to be looked at that are not necessarily the concerns of other agencies. I mean, I think the aviation industry is the one that has brought forward the -- as the example most often and, just an example of how the nuclear industry is different, redundancy, as I understand it, in the nuclear industry often means two identical things that can fail whereas airplanes never run with -- one way of achieving redundancy is two different implementations and the FAA said, well, you know, it would never occur to us to run an airplane with a jet on one side and a propeller on the other. We don't have that same set of or culture of implementing redundancy. So there are some very special things that need to be addressed as digital technology is implemented.

One of the things that came up during the ACRS briefing that I probably should address is our committee did not suggest that the staff loosen its rules in any way for digital technology. We, in fact, endorse the normal and conventional way that 10 CFR 50.59 has been applied. One of our members was a former commissioner, Jim Curtis, and we spent a lot of time trying to understand what the normal process was and stressed that we didn't think digital technology should require a change in that process. So we affirmed essentially how things are done now and suggested that no change be made.

CHAIRMAN JACKSON: Commissioner Rogers?

COMMISSIONER ROGERS: No questions.

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: On that last point, that is not the way the staff interpreted the recommendation and in their response they thought, based on the document that they have submitted to the Commission and we have now put out for public comment that you were suggesting that small changes, which is the heart of the debate over whether we ever endorsed INSAC 125 or we didn't and the staff didn't, that the small changes in safety are going to get there, to our end-reviewed safety question or not.

Small changes, in the view of the staff, is an unreviewed safety question and so they did reject that part

of your recommendation, you know. I know that there is probably debate. We are going to have it in the comments on the 50.59 paper. But I think where the staff has been for some time is that they did not endorse INSAC 125 over this fundamental issue.

MS. MITCHELL: Again, I think that we were very careful to say that what was intended here was that digital technology shouldn't be treated in any way that was different than previous technology. And that just because it had software or hardware that it was automatically an unreviewed safety question was not something that our committee endorsed.

My understanding was that the agency, in terms of these generic letters, has had several drafts of these letters and so there wasn't just one stand on this.

COMMISSIONER McGAFFIGAN: Could I ask a second question that goes to what are the implications of this report for the research program of NRC as opposed to our rulemaking or reg guide efforts? Is there additional research or different research than what we are currently doing in this area?

MS. MITCHELL: The committee made recommendations for action as well as recommendations for how research could proceed or be improved in each of the six technical areas and two strategic areas, so we had some very specific

recommendations.

COMMISSIONER McGAFFIGAN: How large -- a question the Chairman doesn't want me to ask --

CHAIRMAN JACKSON: No, no, no --

COMMISSIONER McGAFFIGAN: What order of magnitude, what order of magnitude research program that we are not currently conducting or reorientation of a current program that, you know, are we talking $5 million per year? Did you get into that level of detail?

MS. MITCHELL: We didn't get into a specific number but I, as a committee member, tried very hard to prevent my fellow committee members from taking unresolved research issues or even technical issues and dumping them in the category of this needs research and this needs dollars before we can continue.

So we tried to suggest directions that could be pursued in light of where things were and where things were likely to be.

CHAIRMAN JACKSON: I just wish to point out for the record that my fellow commissioner and I are actually in concurrence. I am always interested in what the net net dollar amount is but, having spent my career doing research, I know it is very important to define what the problem is, what the research is you want to do, what scope makes sense and what dollars it would take to accomplish that scope and

then one decides on the strategy of how to parse those dollars.

I think in the end what would constitute the right program and what it would cost is something we are interested in.

DR. BOULETTE: It does raise a point I was going to mention in closing and I may as well bring it up now. This committee is unique in its ability to or in its focus in looking at the broad scope of the research program and trying to help the director to prioritize his efforts or the efforts of the staff.

These questions come up and I've got a note in the back of this folder that says the next meeting we have, we have got to talk about shutdown research because it is clearly an area that is significant. As a licensee, I know that. There has been a lot of effort in the industry to try to respond to that concern. As we respond to it, it is very clear that this is a different game, shutdown operation.

So I am sure that Dave and this committee will talk about that over the next several months and try to bring some plan to this.

MR. MAYO: May I make a statement?

I would like to add to Christine Mitchell's statements. I certainly believe there is additional research to be performed. Since I have been on this

committee for several years, we have been saying, well, we are not doing much right now because we are waiting on the study. I have read the recommendations and the report and I believe there is a lot of substance to them and it is something that our subcommittee will be picking up at the next meeting.

CHAIRMAN JACKSON: Okay.

DR. BOULETTE: The next area to be presented to you is in the area of accident analyses and George Bankoff will do that.

George, go ahead.

DR. BANKOFF: In connection with this general idea of longer range view for this committee, I have condensed this report to just three bullets and I welcome comments. There is a lot of meat here and I would like to go over them in just a little detail.

The first thing has to do with the recent development due to a rather extensive study spearheaded by Professor Theophonus at Santa Barbara who, for which he has just received the Ernest Lawrence award from DOE on the strong likelihood of lower head integrity which means, basically, that if you have a reactor with a flooded cavity, if you have that type of design such that you can flood the bottom half of the reactor, that boiling heat transfer will prevent -- will be sufficient to prevent the failure of the

reactor, the retention in core of the melt, the core melt.

That is of such significance, obviously, that it is worth examining it more fully in the research area and justifying further work, possibly. The basic correlations have been shown to exist for various scales of the reactor and it is very simply a function only of the angle, the polar angle of the position. So in view of this, we are recommending that this be examined and maybe reallocate some money.

Now, what this means, basically, is that some reactors such as the AP 600 do have floodable cavities. That is an important thing right there. Some existing reactors also have this. Others, many others do not and so the existing program, which is part of a very large program internationally, should be continued because this does not apply to them. But it is an opportunity for the United States to lead in this area, become again a leader in severe accident technology.

The second bullet has to do with the existing codes and the current scaling methodology. I was very pleased to have a chance, and under the initiative with better cooperation between ACRS and our committee to act as an observer and a participant in the Thermal Hydraulics Committee meetings and as a result of that, I had some rather -- some severe concerns about the current scaling

methodology which I think should not impede in any way the existing process for licensing of AP 600, that is far gone and so forth. But I think that it is time, this methodology is 15 years old, it has never been really examined impartially and objectively and that it is a long-range subject for study, worthwhile, that this is a suggestion.

Finally, this -- the combining of four major codes. We have a code update program. And combining that into a single modern code is clearly a worthwhile idea but it needs to be done quite cautiously. There has been a lot of experience and money invested in the present codes, they function reasonably well. What we want to make sure is that we do it cautiously, that we don't degrade capabilities at the same time as we add to convenience.

CHAIRMAN JACKSON: Well, I guess the question I have is, is this a generalized caution or are there some specific concerns in terms of the approaches being taken or contemplated that are problematic?

DR. BANKOFF: Well, there are some features such as the introduction of transport equations for interfacial area which in principle are interesting but the existing correlations, the existing data in general do not involve interfacial area and so the question is what the database would be when one transfers that into a complex plan.

There is a desire to simplify the codes in the

sense that they would no longer have more than two fields. This may or may not be -- this is a goal that had been expressed from the beginning but it may not be achievable without severe loss of accuracy.

They are talking about also maintaining existing integral capabilities and that is also worthwhile but it is necessary to really have a cost/benefit analysis, because those are expensive, to decide what are the gaps in our knowledge that are really important and will these proposed experiments fill those gaps.

CHAIRMAN JACKSON: Commissioner Rogers?

COMMISSIONER ROGERS: Yes. It was just on this question of experimental validation of final results.

Do you think that there are existing facilities in the world that can provide the data that would be needed to validate a master code of this type?

DR. BANKOFF: Well, I think there are lots of data that has been used to validate existing codes and the question is whether the new code would handle those data as well. We don't have to necessarily get new data. What we have to be able to do is to show that the new code will have the breadth of capability and the accuracy as well for a complex plant, because it is a very -- it can do very well one place and fail miserably in another.

CHAIRMAN JACKSON: My take is that there is a

subtlety to Commissioner Rogers's question, if I may. Because, presumably, the idea of developing this large master code is meant to address certain vulnerabilities or holes in the existing disparatized codes. If that is the case, then, you know, there is a separate issue of modeling the regions and thermal hydraulic space that can be modeled with existing codes versus going and addressing regions that are not addressed.

I am not a thermal hydraullics expert and I think the question, at least the way I would take it, would be are there existing experimental capabilities around the world that would allow one to have some appreciation for the ability of the larger code being contemplated to in fact give information in regions that the current codes do not?

DR. BANKOFF: Well, let me answer by saying I think the major -- a major consideration in combining these codes is maintainability and to reduce the cost of keeping four codes up to one. That is a major consideration. Then the question is, what about all these facilities that have been used in the past? We have facilities, for AP 600 there is an Italian facility, there is a Japanese facility. We have one at Oregon State, we have something at Purdue.

So all of these facilities, all they do is take money.

COMMISSIONER ROGERS: They are all based on some

kind of a single aspect of the system, either modeling full height at the spec facility or modeling something else at the Rosa Facility or trying to model everything else, everything at the Oregon State facility at a quarter scale. So there are scaling questions that are involved with every single one of those facilities and now we are talking about a master code that we hope to be able to rely on but in the long run, the question really comes to something like what the Chairman has said.

Are we going to wind up with the need, really, to validate something in addition to whatever data -- provide data in addition to whatever is there?

DR. BANKOFF: The point is that any, any code that is really good and that has been developed for this kind of data should predict data from any one of these. It should not be limited. You should be able to go back, not only that to the integral scale test but you should be able to look at separate effects tests, smaller scale. It should be code which is quite general. That is the hope of it.

Now, the reason we think about caution is that it never worked out that you can make it that general, that it works very well here but doesn't do so well in some other places.

So when we say it has to be done cautiously, it has to be done with continuous checking to make sure that

you haven't lost something at the same time as you are gaining something.

DR. BOULETTE: Isn't it also true, though, George, that the data set that is being derived for a specific code is derived with that code in mind if you want so it has limitations?

CHAIRMAN JACKSON: Right, that's what we are talking about.

COMMISSIONER ROGERS: That is what we are talking about.

DR. BOULETTE: And my answer would be there would have to be some verification.

COMMISSIONER ROGERS: And my impression is that in every one of these experiments you can get pretty good results if you adjust certain parameters. But then you readjust those parameters when you look at another experiment and that is not a master code; that is something else.

DR. BANKOFF: That has been the situation now. Blind experiments in advance are -- I mean, blind predictions in advance are very difficult.

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: Just one point of clarification. The advice you are giving us at the moment sounds very similar to advice Dr. Caton gave us last fall

when he was looking on behalf of ACRS.

Is that right? You participated with him and do you agree with Ivan Caton's --

DR. BANKOFF: On the codes?

COMMISSIONER McGAFFIGAN: On the codes.

He had some of the same concerns about --

DR. BANKOFF: I didn't go to that meeting so I can't really say.

CHAIRMAN JACKSON: I think we should move on.

DR. BOULETTE: Very good. The next subject is materials and engineering and in this case it is Sumio Yukawa.

DR. YUKAWA: The scope of this subcommittee is to do research that helps support maintenance and control of pressure and structural integrity of the whole pressure boundary system and, as such, it includes materials, engineering and performance evaluation of components and items that primarily constitute the first line of defense in this defense in depth strategy. So it is items like the reactor pressure vessel, piping, valves and so on.

This research area has been an area that involves maturing technology, by and large, as exemplified by big programs like the Heavy Section Steel Technology Program that has been in existence for about 25 years now, the Piping Integrity Program and several other rather large

programs.

The question comes up, well, have we learned enough? And especially in these days of decreasing resources.

Yet we feel, yes, there is a need for selective research because newer issues and needs are coming along, particularly in the areas of less conservative regulations, license renewal issues and, as we have mentioned here earlier, databases for PRA.

So there is a need, we feel, to have research programs in these selective areas and in this context I would like to say that research staff has scheduled a peer review of the whole reactor pressure vessel integrity program for early July and I don't know what the outcome of that will be but it certainly will be some of these questions and issues will be covered there.

We suggest, perhaps, that there ought to be similar critical reviews of other program areas, depending on what the results of this peer review are.

Now, on the next bullet, the third bullet, the third item, I think you have received a letter already which was prompted by a question about well are there simpler or easier ways to measure some of these degradations and properties that accompanies thermal and radiation damage and so forth and I think the reply you received was pretty much,

well, there is very little hope for that right now in the near future.

Given that, we think that basic research to improve mechanistic understanding of the processes that underline engineering performance still needs continuing support.

Then on the fourth and last item, in the direction setting issues, DSI 22, it suggested that opportunities for the three C's, I call them three C's in research, coordination, cooperation, collaboration with industry and international programs and to that I would like to add perhaps that the Naval Reactors Program ought to be somehow or another included. Now there is a lot of questions about that but my impression is that the Naval Reactors Program is now releasing a lot of their at least research study results.

One in particular that I am familiar with has to do with a chemical species diffusion model that really helps to understand what the role of fatigue crack growth in a light water reactor environment is. If we had known about it or this information -- we, I mean, in particular the Boiler and Pressure Vessel Code, if we had known this information we could have done some things differently in the code and presumably it would affect the research program also.

CHAIRMAN JACKSON: So it is called the Chemical Species what?

DR. YUKAWA: This is a diffusion model for specific chemical species in the water that has put an impact on whether or not fatigue crack growth is aided and abetted by the light water reactor environments or not and that is a very, very interesting issue and more than interesting it can be used in defining when the problem is there and when it is not there.

Now, so I just mentioned this about the Naval Reactors Program. I leave it up to somebody more than myself to try to see what can be done there.

CHAIRMAN JACKSON: So you are saying there are perhaps some opportunities in our own yard?

DR. YUKAWA: Yes, I think there is. Because after all, they are operating the same systems that we are and many of the same materials and the same engineering problems.

CHAIRMAN JACKSON: Okay, Commissioner Rogers?

COMMISSIONER ROGERS: Nothing.

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: I do think that's a worthwhile suggestion to follow up.

CHAIRMAN JACKSON: That's right, exactly.

COMMISSIONER McGAFFIGAN: I wonder if we could ask

Dr. Morrison if he has had any chance to look into that.

DR. MORRISON: I haven't had a chance to look into that specific recommendation that Sumio has made. But we do maintain a continuing relationship with the Naval Reactors program and will put that specific item on the table.

COMMISSIONER McGAFFIGAN: Do you get a chance to review the Naval Reactors Research Program and have some visibility into it or is it invisible?

DR. MORRISON: No, it is more picking up instances like this when we get involved in it that we can tie into a specific request. We don't have a broad interaction with Naval Reactors.

CHAIRMAN JACKSON: A lot of the stuff is not generally available.

COMMISSIONER McGAFFIGAN: I understand. I have always felt that Naval Reactors erred on the side of -- too far on the side of keeping everything --

CHAIRMAN JACKSON: Right. I mean, I think the point is made that there is opportunity there and I think that's the point.

DR. MORRISON: Certainly on a very generic issue like this. There are as many differences as there are similarities between the Naval reactors and the light water reactors that we use.

CHAIRMAN JACKSON: Right. And perhaps we can be

more aggressive in pursing these avenues.

DR. MORRISON: Right.

CHAIRMAN JACKSON: That's your point, I think. Okay.

DR. BOULETTE: And that's a comment, again, that we were going to make in a broader scope, not only the Naval Research Program but other initiatives with the industry, conceivably.

The next subject that we wanted to discuss is entitled Methodology of Core Research Capabilities Definition. This was going to be presented to you by John Taylor. I think some of you know John. He is a retired executive with EPRI.

John called in yesterday with the flu. I volunteered to do his presentation. I have also acquired his flu so we will see what happens.

What I thought I might do is read his words. I can do this in about a minute, minute-and-a-half, I think, and hopefully it will stimulate some questions. If it does, I will invite the members of the committee to help me out with the questions.

John says that the methodology which research has developed to define core capabilities is systematic and thorough and should provide an objective assessment of core research capability requirements. The five-step approach is

appropriate. The definition of what constitutes a core research capability, identification of the research functions where support from a core research capability is needed, development of criteria to indicate the amount of support needed for each regulatory function and the importance of that support to the regulatory mission of the agency, documentation of the staff and contract resources needed for each core capability as derived from the first three steps and identification about areas of research that needs to be assessed for core capability.

The Office of Research is to be commended for their efforts as they develop the methodology to obtain the viewpoints of the NRC user offices, NRC program managers and the national labs, deans of nuclear engineering of six universities and industry personnel involved in nuclear research. The following suggestions are made which the committee judges will enhance the results of application of the methodology.

First, 39 areas of research have been identified, primarily in terms of technical skills, where the potential need for core capabilities will be assessed. To provide a clearer basis for the prioritization of these needs, it would be appropriate to define the Office of Research's R&D objectives as well as the technical skills, where are we going, what are we trying to accomplish?

The methodology provides a detailed form of prioritization for assessing for each skill area the regulatory needs which would be fulfilled in that area. Yet, review of the two examples of application of the methodology shows a relatively small difference in capability requirements between an area of high activity, work load driven, and one which is relatively inactive, expertise-driven.

In the planned application of the methodology existing research core capabilities that derived only from the staff of the Office of Research, the committee believes that NRR staff should also be considered as contributing to core capabilities where they have appropriate skills.

From the two examples of application of the methodology, it appears that less important areas will be assigned a minimum of one staff member, a full-time equivalent staff member. This may impose a higher staff requirement than funding permits. Consideration should be given to providing all the needed capability in such areas through contractors, particularly the national labs.

The planned scope of the evaluation that is limited to the current understanding of the regulatory environment does not consider potential future needs and we heard of one this morning in terms of shutdown technology and some research that might be useful and applicable in

that area.

This restriction inhibits planning for new initiatives, particularly in anticipatory research. Lead times in developing new skills can be lengthy.

Although the implementation of the core capability program logically follows the completion of the assessment and Commission approval of core capability needs, preliminary planning should be defined as to how these needs will be maintained or remedied. The implementation will be difficult because of the present and continuing budget restraints and further guidance can come on priorities by assessing the specific difficulties and costs of maintaining capabilities in each area.

This capability assessment is key to maintaining the necessary research competence to permit the Office of Research to meet its responsibilities. Accordingly, it is being given in-depth and high-priority attention by the manager of the Office of Research.

The above comments are intended, on the one hand, to help meet the capability requirements in a limited resource context and on the other hand to enlarge the assessment to include anticipatory research needs.

Those would have been John's comments. Are there any questions or comments to that?

CHAIRMAN JACKSON: Commissioner Rogers?

COMMISSIONER ROGERS: No.

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: No.

DR. BOULETTE: Let me take a few minutes to close and I will be very brief. I had two points that I wanted to make.

One focused on the role of research in the NRC. As you know, the ACRS has already presented its report to Congress and I won't repeat some of those things. But I should say this committee endorses those comments made by the ACRS. We strongly feel that there is a need for continuing and maintaining research in supporting the regulatory process. We are concerned, however, that research is primarily user need driven and that probably the Office of Research ought to try to balance its resources, as tight as they may be, to allow for some preemptive or some exploratory research and we have had discussions with Dr. Morrison about that.

The other point that we would make, and it is highlighting a point that Sumio made in terms of collaboration with the Naval Research Program, we do believe and we do want to encourage the Office of Research to be as collaborative as it can be with the industry and, in particular, for example, the issue of shutdown technology and the research that might support regulatory processes in

that sphere. It would seem to me that working with the industry collaboratively would really help there.

A point that has come up on occasions within our committee and I think with the Commission is the effectiveness of this committee. We have struggled with that for a couple of meetings now because it is a fairly subjective question. Some of the things that we hope to do to assess our effectiveness is to be more diligent about following up on the recommendations and concerns that we expressed in our reports to Dr. Morrison. So you will see in future reports from us a bringing back of older issues that we have raised and how they have been disposed of, how they have been addressed.

Hopefully, from that kind of review, we will be able to assess how effective we have been and how much we have been able to help shape the program of research.

With that, I would say that constitutes our report.

CHAIRMAN JACKSON: Commissioner Rogers?

COMMISSIONER ROGERS: I have nothing.

CHAIRMAN JACKSON: Well, I would like to thank you, Dr. Boulette, members of the committee, and Dr. Morrison, for the briefing. It has been very interesting.

Echoing your words, our research program has to

provide a strong and independent technical capability to undergird our regulatory programs and so the Commission appreciates the committee's efforts in this regard. We would urge you to continue to work with the staff to resolve issues and concerns.

I want to highlight again the area of human factors and because operational experience has shown and you have that experience that human performance is a major factor in the safe operation of nuclear plants and, as we have been talking about, the staff is developing for review an agency wide program plan for human reliability assessment and human performance evaluation. It is expected to be available by the end of June. I think it would be useful for your committee to review the plan, particularly from the point of view of its implications for research and to provide your views to the Commission through the Director of the Office of Research on the adequacy of the plan to advance the state of the art.

DR. BOULETTE: We will do that.

CHAIRMAN JACKSON: And let me just tell you some particular things that I think are important to look at and those have to do with the ability to assess errors of commission, cognitive errors, crew performance, human/machine interface effects and is effect upon performance, information technology effects and that comes

into -- that plays into the digital INC arena, as well as relevant social and organization effects on human performance.

I think if you can provide value-added in that arena and to report those views to the office director and, through him to the Commission, I think that we have talked about the need for effective research in these areas, a well scoped out program. But I believe that scope -- cost follows scope but you have to cost it out and I think, Dr. Morrison, you have gotten some clear indication that there is interest in these areas and I think we should also take to heart what came out of the discussion with Dr. Yukawa relative to looking close at hand for some additional data and research cooperation.

Unless there are any additional remarks by my colleagues, we are adjourned.

[Whereupon, at 12:03 p.m., the meeting was concluded.]

CERTIFICATE

This is to certify that the attached description of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: MEETING WITH NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC) - PUBLIC MEETING

PLACE OF MEETING: Rockville, Maryland

DATE OF MEETING: Friday, May 2, 1997

was held as herein appears, is a true and accurate record of the meeting, and that this is the original transcript thereof taken stenographically by me, thereafter reduced to typewriting by me or under the direction of the court reporting company

Transcriber: Christopher Cutchall

Reporter: Mark Mahoney

BRIEFING PACKAGE

MEETING WITH THE NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC)

MAY 2, 1997

CONTENTS:

- Scheduling Notes
- NSRRC Member List
- NSRRC Charter
- Presentation Slides
- NSRRC Report

SCHEDULING NOTES

Title: Meeting with the Nuclear Safety Research Review Committee (NSRRC)

Scheduled: 10:30 a.m. Friday, May 2, 1997 (PUBLIC)

Duration: Approximately 1-1/2 hours

Participants: Members of the NSRRC

David L. Morrison, Director
Office of Nuclear Regulatory Research

APRIL 1997

NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC)

Dr. E. T. Boulette, NSRRC Chairman
Sr. Vice-President, Nuclear Operations
and Station Director, Pilgrim Station
Boston Edison Co.

Dr. S. George Bankoff
Professor of Chemical and Mechanical Engineering (Emeritus)
Northwestern University

Professor Michael W. Golay
Professor of Nuclear Engineering
M. I. T.

Professor Charles Mayo
Associate Professor of Nuclear Engineering
and Director, Nuclear Reactor Program
North Carolina State University

Professor Christine M. Mitchell
Professor, School of Industrial and Systems Engineering
Center for Human-Machine Systems Research
Georgia Institute of Technology

Mr. John Taylor
Vice President, EPRI (retired)

Dr. Sumio Yukawa
Consultant (metallic materials, components)

NUCLEAR REGULATORY COMMISSION

CHARTER

NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE

1. <u>Committee's Official Designation</u>

NRC Nuclear Safety Research Review Committee (NSRRC)

2. <u>Committee's Objectives, Scope of Activities, and Duties</u>

On a continuing basis, NSRRC will provide advice to the Director of the Office of Nuclear Regulatory Research and through him the Commission, on matters of overall management importance in the direction of the NRC's program of nuclear safety research. Matters requiring NSRRC's attention will be posed by the Commission by the Director of the Research Office, or as an outcome of prior NSRRC deliberations. Nuclear safety research is understood to encompass technical investigations of the implications for public health and safety of the peaceful uses of atomic energy and the reduction of those investigations to regulatory practice.

NSRRC activities will include assessment of and recommendations concerning:

a. Conformance of the NRC nuclear safety research program to the NRC Philosophy of Nuclear Regulatory Research, as stated in the Commission's Strategic Plan, and to specific Commission directions.

b. Likelihood of the program meeting the needs of the users of research.

c. Appropriateness of the longer range research programs and the correctness of their direction.

d. Whether the best people are doing the work at the best places; whether there are other options, including cooperative programs, that would yield higher quality work, or otherwise improve program efficiency.

e. Whether the program is free of obvious bias, and whether the research products have been given adequate, unbiased peer review.

In addition, NSRRC will conduct specialized studies when requested by the Commission or the Director of the Office of Nuclear Regulatory Research. If appropriate, these studies will be published as reports.

3. **Time Period Necessary for the Commission to Carry Out its Purpose**

   In view of the goals and purposes of the Committee, it is expected to be continuing in nature.

4. **Official to whom this Committee Reports**

   The Director of the Office of Nuclear Regulatory Research and, as appropriate, through the Director to the Commission.

5. **Agency Responsible for Providing Necessary Support for this Committee**

   Nuclear Regulatory Commission. Within the Commission, support will be furnished by the Office of Nuclear Regulatory Research.

6. **Description of Duties for which the Committee is Responsible**

   The duties of the NSRRC are solely advisory and are stated in paragraph 2, abcve.

7. **Estimated Annual Operating Costs in Dollars and Man-Years**

   $185,000; 0.8 person-year.

8. **Estimated Number and Frequency of Committee Meetings**

   The Committee will meet at such times and places as it deems necessary, but not less than once a year. Subcommittees may meet as deemed necessary to achieve their assigned tasks.

9. **Committee's Termination Date**

   Two years from the filing date, subject to renewal by the Commission. See also, paragraph 3 above.

10. **Members**

    a. Committee members, including the Chairperson, shall be appointed by the Commission following nomination by the Director of the Office of Nuclear Regulatory Research.

    b. Approximate number of Committee members: 9 to 12.

    c. Members will be chosen to ensure an appropriately balanced representation of the research management community, taking into account: (1) demonstrated experience in high-level management of programs in

applied research; (2) demonstrated expertise in one or more disciplines of applied science and engineering; (3) broad acquaintance with the public health and safety issues associated with the peaceful uses of atomic energy, and (4) a balance of experience in the academic, industrial, and national and not-for-profit laboratory environments.

11. **Date of Filing**: [illegible]

[illegible]
Andrew L. Bates
Advisory Committee Management Officer

PRESENTATION SLIDES

NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE
(NSRRC)

MEETING WITH THE COMMISSION

MAY 2, 1997

INTRODUCTION

- Current Membership
- Subcommittee Make-up
- Schedule
- ACRS Interface

JOINT MEETING OF THE NSRRC I&C AND HUMAN FACTORS/PRA SUBCOMMITTEES

- RES projects in HF and HRA are largely unrelated
- HRA program has limited scope - Atheana had organizational factors
- User needs do not address development of HRA portion of PRA
- Analysis of operating experience should be a resource for relating HF, HRA and PRA
- Longer term programs are needed to advance HRA using HF/HRA projects and PRA analysis of operating experience

PRA IN RISK INFORMED PERFORMANCE BASED REGULATION

- RIPBR is absent at the NRC regional/licensee interface
- Greater use of PRA is needed in guiding regulation
- NRC is advancing dual regulatory emphases, causing confusion
- AEOD could make more use of PRA in helping RES to refine PRA/HRA capabilities
- Many problems of RIPBR implementation are being deferred too long

NAS REPORT ON DIGITAL I&C

- o State-of-the-art technology, widely used
- o Potential to enhance safety and reliability
- o Must be implemented with care in nuclear

ACCIDENT ANALYSIS

- Strong likelihood of in-vessel core retention in reactor with flooded cavity, owing to excellent lower-head cooling
- Existing codes not used for phenomena ranking in current scaling methodology
- Combining four major codes into a single modern code needs to be done cautiously

MATERIALS AND ENGINEERING

- o Scope addresses maintenance/control of pressure and structural integrity of systems
- o Less conservative regulations, license renewal rule, and PRA databases require further research
- o Need to develop mechanistic understanding of damage functions
- o Explore coordination/cooperation with selected naval reactor research programs

METHODOLOGY OF CORE RESEARCH CAPABILITIES DEFINITION

- Methodology systematic and thorough; provides an objective assessment
- To ascertain priorities, R&D objectives as well as technical disciplines should be considered
- Scope limited to RES capabilities is too restrictive
- Consider filling inactive capabilities outside NRC staff
- Scope based only on current activities restricts planning horizon
- Need to identify plans to maintain and remedy core capability requirements

CLOSING

- Role of the Office of Research in NRC
- Effectiveness of NSRRC

NSRRC COMMITTEE REPORT OF ITS APRIL 3-4, 1997 MEETING

Boston Edison

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

E. T. Boulette, PhD
Senior Vice President — Nuclear

April 25, 1997

Dr. David Morrison
USNRC, Office of Nuclear Regulatory Research
Mail Stop TD5, Washington, DC 20555

Dear Dr. Morrison:

Enclosed is the report of the NSRRC Committee meeting on April 3 & 4, 1997. Please note that the subcommittee reports are attached.

If you have any questions on the contents, please call me.

Sincerely,

E.T. Boulette

E. Thomas Boulette, PhD

CC: NSRRC Committee Members
J. Cortez

Encs.
:vep

NSRRC REPORT
APRIL 3-4, 1997

The Nuclear Safety Research Review Committee met on April 3-4, 1997 at Two White Flint North Bldg., Rockville, MD. Members in attendance included E.T. Boulette, Chair, S. George Bankoff, Michael Golay, Christine Mitchell, John Taylor and Sumio Yukawa. Also attending the meeting were David Morrison, RES Director, Jose Cortez, DFO, and several members of the RES staff. A copy of the meeting agenda is attached.

Boulette opened the meeting by reviewing the agenda and emphasizing the need for preparing for the meeting with the commission on May 2. The committee briefly discussed the format for that meeting, with the position taken that each member would make a brief report in their areas of review.

Morrison then made a few remarks regarding the recent management changes in the NRC, with emphasis on the RES organization. Morrison also discussed the evolutions in the NRC's strategic plan emphasizing anticipated changes expected in the next year or so.

Golay, with the assistance of Mitchell, next made a presentation of the joint meeting of the PRA/I&C and Human Factors Subcommittees held on January 24, 1997. The report was reviewed and commented upon, with some discussion of the subcommittees' response to the commission's questions in this area posed last September, 1996. The major observations and recommendations are summarized as follows, with the complete report attached.

The current RES human factors (HF) and human reliability analysis (HRA) programs are largely unrelated. The current HRA programs also are not coupled by work of AEOD in assessing actual event experiences. That situation should change; reorganization would be helpful. The HF/HRA programs of RES are too much dominated by User Needs, uncoordinated, short term-oriented and therefore too often lacking in genuine research content, and typically not organized to formulate and resolve hypotheses.

The HRA program is underfunded and optimistic. It consists of only two projects: ATHENA - for errors of commission, and a project concerning Organizational Factors.

Currently HRA in PRAs relies upon empirical data for quantifying the reliability of simple human actions. With current capabilities HRA is not ready to quantify the reliability of commission errors, cognitive errors, crew performance, human-machine interface effects upon human performance, information technology effects and social and organizational effects upon human performance.

Reorganization of current efforts can be valuable. Important foci should include coupling of HF and HRA efforts, linking HRA/PRA advancement to work of AEOD and NRR, and cooperation with the complementary research programs of other organizations.

The next area discussed by the committee was a report by Mitchell of the NAS final report "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues". Major points of discussion and observations are summarized as follows:

The Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety of the National Research Council submitted its final report to the U.S. Nuclear Regulatory Commission (US NRC) in December 1996. In the report, the Committee made several points. First, the Committee noted that while digital instrumentation and control equipment is state-of-the-art technology, it is widely used both inside and outside the nuclear industry; thus, a large body of experience continues to accumulate. Second, digital technology, including hardware and software, has the potential to enhance safety and increase overall system effectiveness in both nuclear and non-nuclear applications. Finally, due to significant differences between analog and digital technologies, the nuclear industry must implement digital technology with care and caution, particularly in safety-critical systems.

The Committee conducted its analysis in the context of eight key issues: six technical and two strategic. A total of 39 recommendations were made. Technical issues included consideration of characteristics that highlight differences between traditional analog technology and emerging digital systems. Technical issues include systems aspects of distributed systems, software quality assurance, and software reliability. Strategic issues address the regulation process including 10 CFR 50.59 reviews and the adequacy of the US NRC technical infrastructure to respond to quickly emerging technologies.

The Committee reviewed a range of past experience that the US NRC and the nuclear industry have had with digital applications in both current and advanced plants. The Committee endorses the current position of the US NRC and its application of 10 CFR 50.59 with respect to the use of digital technology. The Committee strongly encourages the maintenance of the distinction between digital upgrades that are significant (i.e., pose unreviewed safety questions) and those that are not. Based on this distinction, the Committee recommends that the US NRC tailor the scope and depth of regulatory review in a manner commensurate with significance.

The Committee stressed the importance of maintaining a technical infrastructure which allows the US NRC to address emerging opportunities and issues. The continued change in digital technology and an increase in the rate of change appears inevitable. It is difficult but critical for the US NRC staff to ensure that the vision and technical skills of the staff keep abreast of rapid technology change and potential applications in nuclear applications.

A knowledgeable infrastructure will allow the US NRC to address technical issues in both the regulatory and research areas. The Committee noted that while there are no 'silver bullets' to finally resolve the myriad of technical issues associated with digital

technology, adequate resources and experience exist to permit the incremental incorporation of such technology in nuclear power plants in a manner that ensures the necessary levels of safety and reliability.

Committee deliberations and the revision of the Standard Review Plan proceeded concurrently. At this point, the US NRC staff is reviewing the National Research Council report to ensure that either the Standard Review Plan revision or subsequent deliberations address each of the issues identified in the report.

Professor Bankoff next reported on the Accident Analysis Subcommittee meeting held on April 2, 1997. Several of the subcommittee members were not in attendance for this meeting because of the northeastern blizzard. The principal observations and recommendations of this subcommittee are as follows, with the full report attached.

Coordination with ACRS

George Bankoff attended as an observer and invited participant of the meetings of the ACRS Thermal Hydraulics Subcommittee on December 18-19, 1996 on AP600 PIRT/scaling analysis with Westinghouse and NRR representatives and on February 12-14, 1997 to discuss application of the RELAP code to integral test data obtained in the three integral test facilities (OSU, Italy, Japan).

As a result of these meetings, some apparent difficulties in the current scaling methodology were identified, as described in the letter from Dr. Bankoff to Dr. Catton, dated March 13, 1997, who was the ACRS Subcommittee Chair at the time. It was suggested that this would be an appropriate subject for the study.

The available member of the AA Subcommittee met with RES staff on April 2, 1997 to discuss the severe accident program in the light of recent work by Theofanous, et al., suggesting the strong likelihood of in-vessel core retention for a reactor whose lower head is immersed in a flooded cavity (AP600 and others). There is an active RES program on lower head integrity. The subcommittee wishes to emphasize the following point: the CHF for boiling heat transfer from a submerged lower head has been correlated as a single function of the polar angle, for data from several scales and locations. The minimum CHF is sufficient to keep the lower head cool (a few tens of degrees above the saturation temperature), so that thermal failure, due to creep or local melting, cannot occur. This possibility needs to be thoroughly researched.

Code Updating

The four major codes are slated to be combined into a single modern, modular code. This is a worthwhile project, but needs to be carefully monitored and tested to see whether the technical capabilities of the present codes are being retained.

It is also proposed to convert the fundamental equations (heat, mass and momentum) to more suitable dependent variables, such as interfacial area per unit volume. This should be done very cautiously.

It is also proposed to conduct a continuing experimental program in an NRC-supported facility, such as at Purdue. The subcommittee recommends that prior to running any experiment the gaps in existing knowledge, based on available data, be identified and an analysis be made showing how the experiment will remedy the situation.

Next on the committee's agenda was a briefing by S. Yukawa on the Materials and Engineering Subcommittee meeting also held on April 2, 1997. Attendance for this meeting was also limited by the April 1 storm. Major findings of this subcommittee are as follows with the full text of the report attached.

In a one-half day meeting on April 2, 1997 RES staff presented the status of and plans for research programs in two program areas:

Environmental Qualification (EQ) of Electric Cable Systems
ECCS Strainer Blockage

The presentations indicated that both programs are proceeding on schedule with useful results. Through year 2000, the focus of research on EQ of Cables is on the effect of thermal and radiation aging on the performance of cables used in I&C systems. After that, studies on power cables and connections are planned. The subcommittee's one recommendation for this program is to be sure that the anticipated testing results will be sufficient to fulfill the data needs of PRA methodology for I&C systems.

The testing and experimental phase of the research on Strainer Blockage will be completed in FY 97 and some regulatory actions have already been issued. Additional actions may be issued as needed. This research program has the elements of a useful and beneficial coordination between NRC RES, nuclear industry groups and component suppliers.

Several weeks prior to this committee meeting, an NRC-industry meeting on safety research was held at NRC Headquarters. The meeting took place on March 25th and was attended by utility representation, NRC staff, EPRI, regulatory owners groups, major nuclear vendors, NEI and DOE. The meeting was sponsored by David Morrison of the RES staff, Andrew Kadak of YAEC and John Taylor, member of the NSRRC. Taylor reported the observations coming from the March meeting to the NSRRC and, after some discussion with the committees, compiled the following observations:

RES is to be commended for their efforts, as they developed the methodology to obtain the viewpoints of the NRC user offices, NRC Program Managers in the national labs, deans of nuclear engineering of six universities, and industry personnel involved in nuclear research.

The following suggestions are made which the committee judges will enhance the results of application of the methodology:

Thirty-nine areas of research have been identified, primarily in terms of technical skills, where the potential need for core capabilities will be assessed. To provide a clearer basis for the prioritization of these needs, it would be appropriate to define the RES R&D objectives as well as the technical skills. The methodology provides a detailed form of prioritization by assessing for each skill area the regulatory needs which would be fulfilled in that area. Yet, review of the two examples of application of the methodology shows a relatively small difference in capability requirement between an area of high activity (work-load driven) and one which is relatively inactive (expertise driven).

In the planned application of the methodology, existing research core capabilities are derived only from the RES staff. NRR staff should also be considered as contributing to core capabilities where they have appropriate skills.

From the two examples of application of the methodology, it appears that less important areas will be assigned a minimum of one RES staff member. This may impose a higher staff requirement than funding permits. Consideration should be given to providing all of the needed capability in such areas through contractors, particularly the national labs.

The planned scope of the evaluation is limited to the current understanding of the regulatory environment and does not consider potential future needs. This restriction inhibits planning for new initiatives, particularly in anticipatory research. The lead times in developing new skills can be lengthy.

Although the implementation of the core capability program logically follows the completion of the assessment and Commission approval of core capability needs, preliminary planning should be defined as to how these needs will be maintained or remedied. The implementation will be difficult because of the present and continuing budget restraints and further guidance can come on priorities by assessing the specific difficulties and costs of maintaining capabilities in each area.

This capability assessment is key to retaining the necessary research competence to permit RES to meet its responsibilities. Accordingly, it is being given in-depth and high priority attention by RES management. The above comments are intended on the one hand, to help meet the capability requirements in a limited resource context; but on the other hand, to enlarge the assessment to include anticipatory research needs.

The remainder of the meeting was devoted to a review of the committee's effectiveness in supporting RES and preparation for the meeting with the commission on May 2. With regard to the subcommittee's effectiveness, several observations were made. First, the committee believes it should continue to emphasize a longer term view of research. To that end, the committee needs to spend more attention on the long-term

plans of RES, to ensure appropriate priority is placed in anticipating future needs. An associated concern is related to how much of RES' efforts are devoted to user needs response. Adequate resources need to be reserved for the longer term view, which will require managing the user need responses appropriately.

The committee believes that it can assess its effectiveness by requiring a periodic review of the staff's response to its recommendations. This activity will become a standard agenda item for future committee meeting, with a request that RES formally address all committee recommendations and observations, with a discussion about their resolution.

The committee wants to bring to the commission's attention recent recommendations from the ACRS, the NAS and the NSRRC all supporting continuing safety research. It is considered a vital activity for the NRC, and, in spite of necessary budget reductions, must be managed to ensure maintenance of primary functions.

The final segments of the meeting were devoted to preparations for the meeting with the commission.

Attachment #1

Report of the NSRRC Subcommittees of Probabilistic Risk Assessment and Human Factors Concerning the NRC Human Factors and Human Reliability Analysis Research Programs

INTRODUCTION

The NSRRC Subcommittees of Probabilistic Risk Assessment (PRA) and Human Factors & Instrumentation and Control (HFIC) held a joint meeting on January 24, 1997 to review the progress of Human Factors research and to discuss the Commission's questions in a September 10, 1996 SRM. The purpose of this meeting was to respond to the Commission's request for the NSRRC to:

□ 1) Continue to review the progress of human factors research
□
□ 2) Identify those human factor areas that can be treated adequately in PRA
□
□ 3) Identify human factors areas where progress for inclusion in PRA is likely.
□
□Reference material provided by RES included:
□
□ 1) The December 30, 1996 ACRS Questions on Human Performance Plan addressed to the EDO
□
□ 2) Presentation materials provided by RES for the last meetings of the two
□ subcommittees
□
□ 3) The NSRRC report on the last PRA and HFIC subcommittee meetings
□
□ 4) The August 1995 and July 1996 versions of the Human Performance Program Plan
□
□ 5) The January 13, 1997 Quarterly Status Update for the PRA Implementation.>□
□

HUMAN PERFORMANCE PLAN

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The Human Factors Coordination Committee meets approximately every six months to coordinate the NRC's human factors activities and to update the NRC's Human Factors Program Plan as necessary. Revision 1 to this plan (July, 1996) summarizes the goals, objectives, lead office, and other information supporting this plan.

RES is the lead office in this plan for accomplishing the following goals:

GOAL 1: Assure the Adequate Safety Performance of Nuclear Facility Personnel

Objective 1.1 Assure that nuclear facility personnel are adequately qualified and that staffing is appropriate

- 1.1.3 Modify Regulatory Guide 1.8
- 1.1.6 Develop Performance-based Approach to Determining Fitness-for-Duty
- 1.1.4 Develop Guidance for Staffing at Operating Reactors
- 1.1.8 Develop Guidance for Staffing for Advanced Reactors

Objective 1.2 Assure that nuclear facilities have effective human-system interfaces

- 1.2.4 Develop NUREG-0700, Rev.1
- 1.2.5 Develop Guidance for Advanced Alarm Systems
- 1.2.6 Develop Guidance for Hybrid Control Rooms
- 1.2.7 Develop Guidance for Display Navigation
- 1.2.8 Prioritized Human-System Interface Issues
- 1.2.9 Develop Future Revisions to NUREG-0700
- 1.2.11 Develop Guidance for Computerized Job Performance Aids

Objective 1.3 Assure that nuclear facilities have effective organizational practices

- 1.3.2 Develop Method to Quantify Organizational Performance Factors

GOAL 2: Provide Empirically Based Information to the Regulatory Process

Objective 2.1 Assure that human performance is effectively assessed

- 2.1.3 Revise Human Performance Related Management Directives
- 2.1.4 Revise Human Performance Investigation Process
- 2.1.5 Analyze and Disseminate Human Reliability Assessments / Probabilistic Risk Assessments (HRA/PRA Information

Objective 2.2 Assure the availability and use of adequate human performance information

- 2.2.10 Determine the Feasibility of Using Task Network Modeling for Regulatory Applications

□ 2.2.11 Develop Methods to Obtain Empirical Operational Data
□
□ Objective 2.3 Assure the use of empirically-based HRA in PRA as appropriate
□
□ 2.3.1 Develop Methods for Quantifying Errors of Commission
□ 2.3.2 Develop Methods for Quantifying Organizational Factors for PRAs
□
□ Objective 2.4 Assure that research is focused is based on providing technical bases for Commission policies and regulatory decisions
□
□ 2.4.4 Develop Standardized Format for User Needs
□ 2.4.7 Conduct Feasibility Studies of Emergent Issues
□
□GOAL 3: Ensure Adequate Availability and Effective Coordination of NRC Resources to carry out the Human Performance Programs.
□
□ Objective 3.2 Assure effective communication within and across offices
□
□ 3.2.2 Develop and Implement Human Performance Orientation Training
□
□ Under Objective 2.2 Assure the availability and use of adequate human performance information in the Human Performance Program Plan, AEOD has the following six activities:
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□ 2.2.1 Integrate Human Performance Information Into a Consolidated Event Data Base
□ (Schedule TBD)
□
□ 2.2.2 Modify the SCSS to Include More Human Performance Information and make
□ SCSS More Widely Available
□ (continuing)
□
□ 2.2.4 Make AEOD Human Performance Event Database Available Throughout the
□ Agency
□ (Schedule TBD)
□
□ 2.2.5 Revise NUREG-1022 to Better Define the Human Performance Information
□ Required in Licensee Event Reports
□ (Schedule TBD)
□
□ 2.2.6 Determine Information Needs for HRAs
□ (Schedule TBD)
□
□ 2.2.7 Assess HRA Models to Assure Human Performance Databases Contain

□ Information Useful for HRA Models
□ (Schedule 3Q/FY97).
□

Of these items, only Objective 2.3 is related to the use of HRA in PRA. Human reliability data and such as being collected by the AEOD would appear to be essential to the development of HRA models and their incorporation in PRA.

□

□**PRA**

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□In the January 13, 1997 Quarterly Status Update for the Probabilistic Risk Assessment (PRA) Implementation Plan (SECY-97-009), the Subcommittee found RES referenced as a lead office in the following categories:

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□1.7 Regulatory Effectiveness Evaluation (With NRR)

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Assess the effectiveness of two major safety issue resolution efforts (i.e. SBO and ATWS rules) for reducing risk to public health and safety.

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□1.8 Advanced Reactor Reviews - (with NRR)

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□ Develop independent technical analysis and criteria for evaluating industry initiatives and petitions regarding simplification of emergency preparedness regulations.

□

□1.9 Accident Management (with NRR)

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□ Develop generic and plant specific risk insights to support staff audits of utility accidents management programs at selected plants.

□

□1.10 Evaluating IPE Insights to Determine Necessary Follow-up Activities (with NRR)

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□ Use insights from the staff review of IPEs to identify potential safety, policy, and
□ technical issues, to determine an appropriate course of action to resolve these potential issues, and to identify possible safety enhancements.

□

□2.1 Develop Regulatory Guidelines

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□ Regulatory Guides for industry to use in risk-informed regulation (General, IST, ISI, GQA, TS).

□

□2.2 Technical Support

□

□ Provide technical support to agency users of risk assessment in the form of support for risk-based regulation activities, technical reviews, issue risk

assessments, statistical analyses, and develop guidance for agency uses of risk assessment.

□

□2.3 Support for NRR Standard Reactor PRA Reviews

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□ Modify 10CFR52 and develop guidance on the use of updated PRAs beyond design certification.

□

2.4 Methods Development and Demonstration

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□ Develop and demonstrate methods for including aging effects in PRAs.

□

□ Develop and demonstrate methods for including human errors of commission in PRAs.

□

□ Develop and demonstrate methods to incorporate organizational performance into PRAs.

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□ Develop and demonstrate risk assessment methods appropriate for application to medical and industrial licensee activities.

□

□2.5 IPE and IPEEE Reviews

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□ To evaluate IPE/IPEEE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover vulnerabilities; and to document the significant safety insights resulting from IPE/IPEEEs.

□

□2.6 Generic Issues Program

□ To conduct generic safety issue management activities, including prioritization, resolution, and documentation, for issues relating to currently operating reactors, for advanced reactors as appropriate, and for development or revision of associated regulatory and standards instruments.

□

□3.2 Accident Sequence Precursor Program

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Complete quality assurance of Rev. 2 simplified plant specific models.

□

□ Complete feasibility study for low power and shutdown models.

□

□ Complete initial containment performance and consequence models

□

□3.6 Staff Training (with AEOD)

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☐ Develop and present Appendix C training courses

☐

☐4.1 Validate risk analysis methodology developed to assess most likely failure modes and human performance in the use of industrial and medical radiation devices.

☐ Continue the development of the relative risk methodology, with the addition of
☐ event tree modeling of the brachytherapy remote afterloader.

☐

☐ Extend the application of the methodology and its further development into
☐ additional devices, including teletherapy and the pulsed high dose rate afterloader.

☐

☐4.2 Continue use of risk assessment of allowable radiation releases and doses associated with low-level radioactive waste and residual activity (with NMSS).

☐

☐ Develop decision criteria to support regulatory decision making that incorporates
☐ both deterministic and risk-based engineering judgment.

☐

☐4.3 Develop guidance for the review of risk associated with waste repositories (with NMSS).

☐

☐The only two activities related to the incorporation of HRA in PRA are the 2.4 Methods
☐Development and Demonstration activities of

☐

☐ - Develop and demonstrate methods for including human errors of commission in
☐ PRAs

☐

☐ - Develop and demonstrate methods to incorporate organizational performance
into
☐ PRAs.

OBSERVATIONS OF THE SUBCOMMITTEES

The subcommittees have several areas of consensus concerning the information presented to us. We summarize them in the following discussion.

The RES Projects in the Areas of Human Factors (HF) and Human Reliability Analysis (HRA) are largely unrelated

In the presentations from the RES and NRR staff members concerned with the HF and HRA programs, respectively, it became evident that the great majority of the HF programs have been formulated without regard to the need to improve the understanding of HRA. Rather, most of the work being done is in response to NRR user needs. As such each element is typically short term-focused and not coordinated with the others. The fact that user needs are not formulated in a coherent fashion is sufficient to explain the latter aspect.

This incoherence is also reflected in the apparent lack of interaction by the HF and HRA efforts with the real-world analysis efforts of AEOD. In an effective, comprehensive research program it would be logical to expect a portion of to be comparing knowledge gained in individual projects with what is being learned about actual operating plants, and also contributing to the formulation of the AEOD research plan. Further, a connection between the operating plant analyses and the research program would be expected to provide theoretical models in terms of which to make sense of the observed operating results. The absence of such interactions is symptomatic of the absence of a meaningful HF/HRA research plan within RES.

As most of the current user needs and independent research efforts were formulated prior to the current NRC emphasis upon Risk Informed Performance Based Regulation (RIPBR) it is perhaps not surprising that they are unconcerned with how the work to address these user needs could be useful in the HRA portion of a probabilistic risk assessment (PRA).

However, regardless of the cause of the lack of connection between the HF and HRA programs, it would be valuable to coordinate them. Logically each HF project should have an implication for the understanding of HRA, and should be able to contribute to an increasing body of knowledge in this area. Hopefully this will be done.

The HRA Program is Far Too Small for Likely Success

In addition to the problems mentioned above the HRA program appears to have two main elements: the ATHENA project - intended to provide models and data concerning errors of commission and of cognition and the Organizational Factors (OF) project - intended to provide contributions concerning the effects of organizational structure and management approach upon power plant risks. In fact the information provided to our Subcommittees to-date has been so little that it is impossible to make any meaningful statement concerning the goals or methods of either project. However, we can note that the ATHENA budget

of $700k annually is small in terms of the challenges of developing a deep understanding the problems being addressed; the area of HF is one where the NSRRC has repeatedly recommended either termination of the work of RES or a fresh start, based upon new and better ideas. We have seen no evidence of the latter; so we remain skeptical that the OF project will be useful.

The HF approach within RES is apparently based upon simple empirical handbook models of human error rates, developed about 17 years ago. These are not very useful concerning such topics as digital human-machine interfaces, cognitive errors, crew interactions, psychological motivations of human performance and social and organizational interactions. Progress in these areas is difficult and requires a sustained and substantial research commitment. Such a commitment is not evident in the RES program, which can be characterized as under-funded and optimistic. The two current efforts may have the right goals, but we would expect them to be accompanied by several more projects, at a much more substantial budgetary level in order to have any reasonable chance of making important contributions. Rather, the current approach appears to be one of doing what is feasible under stringent circumstances, and hoping for good luck.

However, it is conceivable that things are much better than this summary indicates. We need much more information from RES before we can say more. Through this letter we request that information.

DISCUSSION OF THE RES PROGRAMS

An essential tool in the current NRC thrust to implement RIPBR is PRA. An area of important PRA uncertainty and weakness is that of HRA (as is recognized among the NRC staff). As is discussed above the current program of RES does not promise to improve this situation. This is because currently RES lacks an effective research program in this area. Rather they have two small projects, and poor linkage of HRA improvement to most of the relevant efforts within the overall HF program. Much of this can be explained by the internal NRC policy of focusing RES's resources upon user needs, upon severe budgetary restrictions and perhaps from too much experience within the research organization of focusing upon short term requirements.

If the NRC is to be realistic about improving the current state of knowledge in HRA it is necessary to establish a long duration (i.e., > 10 years), substantially funded (i.e., > $3 Million annually) program for this purpose. Much of this could be done by reprogramming the combined HF and HRA efforts, but doing so would likely require serving NRR's user needs elsewhere.

The test of whether a revised research program is likely to be effective could be provided by examining the degree to which it would be able to help answer the following questions adequately:

1. What questions must be answered in order to understand HRA sufficiently,

2. How is the overall NRC effort trying to get these answers,
3. How does the work of RES fit into the overall NRC effort,
4. What are the important elements of the RES HRA and HF programs, and how do they conform to the rationale of the overall NRC effort,
5. How are the expected products of these elements planned to be employed?

One should expect the answers to these questions to provide the bases of the HF/HRA research program.

However, we asked these questions in examining the HF and HRA programs and still need much more information before we will know the staff's understanding of them. This situation is symptomatic of the weakness of current efforts in attempting to constitute an effective HF/HRA research program.

Attachment #2

ACCIDENT ANALYSIS SUBCOMMITTEE REPORT

1. Coordination with ACRS

In accordance with the letter from John Larkins to David Morrison, dated December 13, 1996, on ACRS and NSRRC coordination, the Accident Analysis (AA) Subcommittee Chair, George Bankoff, attended the meetings of the ACRS Thermal Hydraulics Subcommittee on December 18-19, 1996 to review AP600 PIRT/scaling analysis with Westinghouse and NRR representatives, and on February 12-14, 1997 to discuss application of the RELAP code to integral test data obtained in the three integral test facilities (OSU, Italy, Japan). A substantial fraction of both meetings was occupied by the application of the current scaling methodology to the small-break accident. This is not the most dangerous one, but exercises the passive features of the AP600 design more fully. For the December meeting Westinghouse provided two major reports:

WCAP-14727, "AP600 SCALING AND PIRT CLOSURE REPORT" Propriety Class 2, and
WCAP-14772, "AP600 TEST PROGRAM OVERVIEW" Non-Proprietary Class 3.

The first was a comprehensive summary of previous work, detailed in a large number of previous reports of the past six years. In itself, it was quite massive, containing more than 190 figures and running about 300 pages. The second was intended as a guide to the first, although there were a number of complaints by the consultants that the necessary information to back up statements made during the presentation could not easily be found. It was requested that a "guide to the guide" be supplied, in the form of an additional summary report pointing to the evidence supporting all assumptions and conclusions.

The plant structure is quite complex, and when added to the changes in behavior in various time periods in a small break accident, the level of detail becomes too much to describe even briefly.

The Phenomena Identification Ranking Table was developed for each type of transient, including LOCA, SBLOCA, operational transients and long-term cooling. The scaling analysis is performed by writing the global conservation equations for mass, momentum and energy at the system level for specific time periods during the transient, and normalizing by dividing through by the coefficient of the driver term in each equation. The coefficient of the driver term, such as the hydrostatic head term in the momentum equation, is then unity, while the coefficients of the other terms can be used to evaluate which terms are of minor importance. In a loop the sum of the pressure drops, including frictional and inertial terms, is equal to the hydrostatic pressure difference. This means that the sum of the dimensionless coefficients should be close to unity if the dimensionless dependent and independent variables are properly scaled. This turns out frequently to be not the case. Other difficulties exist, which are described in the letter from Bankoff to Dr. Catton, dated March 13, 1997, who was the ACRS Subcommittee chair at the time,.

2. The available member of the AA Subcommittee met with RES staff on April 2, 1997 to discuss the severe accident program, in the light of recent work by Theofanous, et al. suggesting the strong likelihood of in-vessel core retention for a reactor whose lower head is immersed in a flooded cavity (AP600 and others). There is an active RES program on lower head integrity, as summarized in the accompanying viewgraphs. The subcommittee wishes to emphasize the following points:

1. The CHF for boiling heat transfer from a submerged lower head as summarized on

pp. 30, 31 of the viewgraphs, is correlated as a single function of the polar angle, θ, where at the bottom of the reactor $\theta=0^\circ$. The minimum CHF is sufficient to keep the lower head cool (a few tens of degrees above the saturation temperature), so that thermal failure, due to creep or local melting, cannot occur.

2. This possibility needs to be thoroughly researched, and the subcommittee recommends switching of some funds from internal heat transfer dynamics in the lower head, as well as ex-vessel accident analysis, to provide an adequate budget. If verified, this scenario means that for floodable cavities the ex-vessel portion of the severe accident scenario is of negligible probability.

3. It is early to make a judgment, but the RASPLAV experiment, being a full-scale melting experiment with electrical heat, is difficult to instrument, expensive, prone to failure, and slow to reconstruct and to run. This may be a place to economize.

3. Code Updating

The four major codes are slated to be combined into a single modern, modular code. This is a worthwhile project, but needs to be carefully monitored and tested. Some simplifications, like going to a two-fluid code, may lead to reduction of capability of handling some types of boiling and dispersed flows, especially with multiple materials. Nevertheless, this is the proper direction to go, but with constant checking, evaluation and reevaluation to see whether the technical capabilities of the present codes are being retained.

It is also proposed to convert the fundamental equations (heat, mass and momentum) to more suitable dependent variables, such as interfacial area per unit volume. This should be done very cautiously, in view of the large body of experiments in which these measurements were not made, together with the correlations based on these experiments.

It is also proposed to conduct a continuing experimental program in an NRC-supported facility, such as at Purdue. The subcommittee recommends that prior to running any experiment, the gaps in existing knowledge, based on available data, be identified, and an analysis be made showing how the experiment will remedy the situation.

Attachment #3

NSRRC SUBCOMMITTEE ON MATERIALS AND ENGINEERING
MINUTES OF APRIL 2, 1997 MEETING

The Subcommittee met in a morning session on this date at the NRC offices in Rockville. Sumio Yukawa, Subcommittee Chair, was the only member able to attend the meeting. E. T. Boulette and John Taylor were unable to attend due to a New England snowstorm and a prior commitment.

The meeting concentrated on presentations and discussions on two research program areas which were:

Environmental Qualification (EQ) of Electric Cable Systems
presented by Michael E. Mayfield, Chief Electrical, Materials, and
Mechanical Engineering Branch
Division of Engineering Technology
ECCS Strainer Blockage Research
presented by Michael L. Marshall, Jr.
Generic Safety Issues Branch
Division of Engineering Technology

The funding for these two program areas together total approximately $1.5 million in FY97. The funding in FY94 was about $2.0 million. The following is a brief summary of the issues, scope, contents, and status of research in these two program areas.

EQ of Electrical Cables

The current research in this area concentrates on cables and associated connectors used in instrumentation and control applications inside the containment. The concern relates to the reliability, capability, and performance after thermal aging and radiation damage effects on the insulations and connections. Testing requirements and qualification criteria for the cables have changed over the last twenty years so that cables in operating plants have been "qualified" to differing requirements. The staff initially identified 43 EQ technical issues deriving from these differing requirements. A literature review effort has reduced these to 19 issues and further resolutions are likely.

The ongoing research program includes studies to understand the differences, if any, between artificial and natural aging on the chemical, electrical, and mechanical characteristics of the cables. These characteristics will be measured on new and artificially aged items. Additionally, cables have been acquired from Yankee Rowe and Trojan plants which had twenty and ten years of service respectively. The testing includes exposure to steam conditions representative of LOCA events. Among the expected results is

verification of aging models that can be used for time extrapolations for license renewal purposes.

The program plans schedule completion of major part of the research on I&C cables by the year 2000. After that, research on power cables and penetrations are planned.

The Subcommittee believes that results of this research are vitally needed. The program is well structured and proceeding on schedule. There is some concern about whether the scope and extent of the program will fulfill the needs for a PRA analysis of the I&C system. This concern should be discussed with those responsible for PRA methodology of the I&C system to determine if the anticipated results are sufficient for their needs. The discussion should include the fact there may be a paucity of and uncertainty in the data between the observed aging deterioration and the functionality of the I&C system which may reduce the value of PRA applications to this problem.

STRAINER BLOCKAGE

The strainer prevents passage of deleterious debris in the water drawn from the wet well and their entrance into pumps, valves, and nozzles of the ECCS system. Recent plant incidents and studies of potential circumstances of a LOCA event indicate that blockage in and damage to the strainers may occur more easily and quickly than anticipated which could reduce the flow required for ECCS conditions. One possible consequence of a LOCA event involves a water jet that can disintegrate and spread fibrous thermal insulation and other materials into the wet well. Test and analysis results point to a very high conditional probability of loss of ECCS capabilities. A computer code to analyze strainer blockage has been developed and is publicly available. To date, the studies have been primarily on BWRS in cooperation with the BWR Owners Group (BWROG) but studies for PWRs are ongoing.

Recent regulatory action has requested BWR licensees to implement appropriate actions during outages starting after January, 1997, to ensure that ECCS functions are not impaired. The NRC has determined that this is a compliance issue, not a safety enhancement issue.

The research schedule is that all of the experiments and tests will be completed in FY97. The currently remaining FY97 work is principally concerned with debris transport. The NRC RES results and plans plus those of BWROG and new design strainers by vendors should bring a closure to this problem.

The Subcommittee believes that this research program contains a beneficial and coordinated mix of inputs from NRC, nuclear industry groups, and component suppliers.

Attachment #4

AGENDA

NUCLEAR SAFETY RESEARCH REVIEW COMMITTEE (NSRRC)

April 3-4, 1997

Room T-10A1. Two White Flint North (TWFN) Building

11545 Rockville Pike. Rockville. MD

Thursday. April 3

9:00 - 9:30	Opening remarks	E. Thomas Boulette. Chairman. NSRRC
	Current status	D. Morrison. Director RES
9:30 - 10:15	Report of the Joint Meeting of the PRA/I&C and Human Factors Subcommittees on their meeting of January 24, 1997	M. Golay. NSRRC
10:15 - 10:45	Report of the NAS final report "Digital Instrumentation and Control Systems in Nuclear Power Plants. Safety and Reliability Issues"	C. Mitchell. NSRRC
10:45 - 11:00	BREAK	
11:00 - 11:30	Report of the Accident Analysis Subcommittee on its meeting of April 2, 1997	S. George Bankoff. NSRRC
11:30 - 12:00	Report of the Materials and Engineering Subcommittee on its meeting of Aril 2, 1997	S. Yukawa. NSRRC
12:00 - 1:00	LUNCH BREAK	
1:00 - 2:00	Industry meeting report on core research competencies	J. Taylor. NSRRC
2:00 - 2:15	BREAK	
2:15 - 5:00	Discussions on Research Core Competencies	NSRRC Committee RES Staff

Friday, April 4

8:30 - 10:45	Continuation of Discussions on Research Core Competencies	NSRRC Committee RES Staff
10:45 - 11:00	BREAK	
11:00 - 12:00	Review of NSRRC Committee Effectiveness	NSRRC Committee RES Staff
12:00 - 2:30	LUNCH BREAK	
2:30 - 5:00	Preparation for May 2. 1997 NSRRC meeting with the Commission	NSRRC Committee
	3:00 - 3:15 BREAK	
5:00 PM	ADJOURN	